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CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

April 27, 2000
CY-00-088

Ref. 49CFR107 & 49CFR173

Research and Special Programs Administration
U.S. Department of Transportation
400 7th Street, SW
Washington, D.C. 20590-0001

Attention: Ms. Sandra Cureton

Reference: (a) Letter, CYAPCO to U.S. DOT, "Request for Exemption for Shipment of the Haddam Neck Nuclear Plant Reactor Vessel," dated March 30, 2000.

SUBJECT: REQUIRED COPIES OF APPLICATION

Dear Ms. Cureton:

In Reference (a), CYAPCO submitted its request for exemption for shipment of the Haddam Neck Nuclear Plant reactor vessel. This letter submits the required two (2) copies of the application which were inadvertently omitted from the original transmittal.

We apologize for this oversite. If additional information is required, please contact Mr. Gerry P. van Noordennen, Manager of Regulatory Affairs, at (860) 267-3938.

Sincerely,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

Russell A. Mellor – Vice President
Operations and Decommissioning

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD • EAST HAMPTON, CT 06424-3099

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March 30, 2000
CY-00-016

Ref. 49CFR107 & 49CFR173

Associate Administrator for Hazardous Materials Safety
Research and Special Programs Administration
U.S. Department of Transportation
400 7th Street, SW
Washington, D.C. 20590-0001

Attention: Exemptions, DHM-3 1

SUBJECT: REQUEST FOR EXEMPTION FOR SHIPMENT OF THE HADDAM NECK NUCLEAR PLANT REACTOR VESSEL

Connecticut Yankee Atomic Power Company (CYAPCO) is in the process of decommissioning the Haddam Neck Plant (HNP). Therefore, we hereby request the necessary exemptions for the shipment of the reactor vessel within a Reactor Vessel Transport System (RVTS) from the HNP to the low-level radioactive waste burial site at Barnwell, South Carolina. The RVTS will provide a package/transport system with an equivalent safety level to that of a DOT Industrial Package Type 2 (IP-2) package. The RVTS will be transported under a DOT exemption pursuant to 49 CFR Part 107.105 via barge to the Savannah River Site (SRS) in Aiken, South Carolina and via land transporter from SRS to Barnwell. This shipment will be exclusive use, one-time only, and will be performed in accordance with the transportation plan as described herein. The current project schedule reflects the shipment to occur after September 1, 2000.

The intact HNP Reactor Pressure Vessel (RPV) containing some reactor internals components and potentially RPV mirror insulation will be packaged and grouted, with low density cellular concrete, within a 3-inch thick steel canister with the RPV head bolted to the exterior of the canister.
The components of the reactor vessel internals exceeding 10 CFR Part 61 Class C limits (Greater Than Class C, GTCC) will be removed and will not be shipped with the RPV. The GTCC will remain at the HNP site for future disposition.

The RVTS consists of the "package" which is used to contain the RPV and its Class 7 (radioactive) materials, an integrated tie-down system for barge and land transport, and a transportation and emergency response plan. A complete description and discussion of the exemptions requested, the RVTS, its transportation plan, and emergency response plan is provided in the following four (4) attachments to this request:

Attachment 1 – Compliance Matrix
Attachment 2 – Transport System Description
Attachment 3 – Reactor Vessel and Internals Characterization
Attachment 4 – Transportation and Emergency Response Plan

The exemptions requested for one-time use of this transport system include that:

1. The package is designated as a non-specification package since the proposed Transport System, under normal conditions of transport and prescribed operating conditions, provides safety equivalence to that of an IP-2 package,

2. The package is exempted from the drop requirement from the orientation that causes ***"maximum damage"***,

3. The package contents is considered LSA III material,

4. The package contents, including the RPV itself, is considered a ***"collection of solid objects"*** and that the requirements of 10 mSv/hr (1R/hr) at 3 meters, as provided in 49 CFR Part 173.427(a)(1), be applied from the unshielded surface of the RPV exterior,

5. The package contents classified as LSA-III materials is exempted from the leachability requirements of 49 CFR Part 173.468, and

6. The package is exempted from the stacking test requirements of 49 CFR Part 173.465(d).

The RVTS will meet all of the other requirements of 49 CFR Part 173.

The bases for the requested exemptions are due to the unique characteristics of the Class 7 (radioactive) material to be transported, the packaging, and the administrative controls that will be implemented during transportation. The justification for each exemption requested is provided in the Compliance Matrix (Attachment 1). The remaining attachments provide additional supporting information.

The Compliance Matrix (Attachment 1) addressing the requirements of 49 CFR Part 107 and Attachments 2 through 3 together demonstrate that the shipment of the HNP reactor vessel can be performed with a level of safety equal to or greater than that of an IP-2 package.

At your convenience, we are prepared to meet with you to discuss this request and respond to any questions. If additional information is required, please contact Mr. Gerry P. van Noordennen, Manager of Regulatory Affairs, at (860) 267-3938.

Sincerely,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

K. Heider for R.A. Mellor

Russell A. Mellor – Vice President
Operations and Decommissioning

Attachments:

Attachment 1 – Compliance Matrix
Attachment 2 – Transport System Description
Attachment 3 – Reactor Vessel and Internals Characterization
Attachment 4 – Transportation and Emergency Response Plan

cc: R. Boyle, DOT, Office of Hazardous Materials Technology
T.L. Fredrichs, Project Manager, USNRC

ATTACHMENT 1

COMPLIANCE MATRIX

ATTACHMENT 1
COMPLIANCE MATRIX

EXEMPTION REQUEST FROM THE PACKAGING REQUIREMENTS OF 49CFR 173 FOR THE SHIPMENT OF THE HADDAM NECK PLANT (HNP) REACTOR VESSEL TRANSPORT SYSTEM

For ease of review and processing, this exemption request was prepared under the guidelines of 49CFR 107.105 in effect as of October 1, 1999.

49CFR 107.105:	Application for exemption.
49CFR 107.105(a):	General
49CFR 107.105(a)(1):	The requested need date for this exemption is **July 31, 2000**

Two copies of this exemption have been delivered to:

Associate Administrator for Hazardous Materials Safety
Research and Special Programs Administration
U.S. Department of Transportation
400 7th Street, SW
Washington, D.C. 20590-0001
Attention: Exemptions, DHM-31

49CFR 107.105(a)(2): The correct applicant name, address and responsible agent for this exemption is:

Applicant:
Connecticut Yankee Atomic Power Company (CYAPCO)
362 Injun Hollow Road
East Hampton, CT 06424-3099
Attention: Mr. G.P van Noordennen
Regulatory Affairs Manager
Telephone: (860) 267-393 8

Contractor:
Bechtel Power Corporation (BPC)
Decommissioning Operations Contractor
Haddam Neck Plant
362 Injun Hollow Road
East Hampton, CT 06424-3099
Attention: Mr. Donald Scribner
Project Engineer
Telephone: (860) 267-3047

ATTACHMENT 1
COMPLIANCE MATRIX

EXEMPTION REQUEST FROM THE PACKAGING REQUIREMENTS OF 49CFR 173 FOR THE SHIPMENT OF THE HADDAM NECK PLANT (HNP) REACTOR VESSEL TRANSPORT SYSTEM

49CFR 107.105(a)(3):	CYAPCO and BPC are United States corporations.
49CFR 107.105(a)(4):	This is not a request for a Manufacturing Exemption.
49CFR 107.105(b):	***Confidential treatment*** Confidential treatment of this exemption is not requested.
49CFR 107.105(c):	***Description of exemption proposal***
49CFR 107.105(c)(1):	With regard to the transportation of one reactor vessel within a Reactor Vessel Transport System (RVTS) from the Connecticut Yankee (CY) site in Haddam Neck, CT to the low level radioactive waste burial site at Barnwell, South Carolina, the Applicant seeks relief from the requirements of 49CFR 173 as follows;

PACKAGING REQUIREMENT

The requirement of 49CFR 173.427(a) that low specific activity (LSA) material must be packaged in accordance with 49CFR 173.427(b) or (c).

DOSE RATE AT 3 METERS

The requirements of 49CFR 173.427(a)(1) regarding the 10 mSv/hr (1 Rem/hr) radiation dose limitation at 3 meters from the unshielded material.

LSA III DEFINITION

The requirements of 49CFR 173.403 regarding the definition of LSA-III material, which does not provide for surface contaminated LSA material.

ATTACHMENT 1
COMPLIANCE MATRIX

EXEMPTION REQUEST FROM THE PACKAGING REQUIREMENTS OF 49CFR 173 FOR THE SHIPMENT OF THE HADDAM NECK PLANT (HNP) REACTOR VESSEL TRANSPORT SYSTEM

LSA III MATERIAL LEACH TESTING

The leach testing required by 49CFR 173.468 for LSA III material which is also included in the definition of LSA III in 49CFR 173.403.

Sec 173.403 Definitions requires that LSA materials consist of Class 7 (radioactive) material with limited specific activity and the determination of such specific activities may not consider the shielding materials surrounding the LSA material. For LSA-III solids, this Section further provides:

1. That such materials meet the requirements of 49CFR 173.468, which provides detailed requirements for the LSA III leach testing and,

2. Have the Class 7 (radioactive) material "distributed throughout a solid or a collection of solid objects", and,

3. Have an average specific activity not to exceed 2x1 0-3 A2/g, and,

4. Consist of Class 7 (radioactive) material which is relatively insoluble so that even under loss of packaging, the loss of material by leaching in water for 7 days shall not exceed a 0.1 A2 quantity.

FREE DROP TEST

The drop orientation requirements of 49CFR 173.465(c) that IP-2 packages must satisfy the requirements of a drop test onto the target so as to suffer maximum damage to the safety features being tested.

ATTACHMENT 1
COMPLIANCE MATRIX

EXEMPTION REQUEST FROM THE PACKAGING REQUIREMENTS OF 49CFR 173 FOR THE SHIPMENT OF THE HADDAM NECK PLANT (HNP) REACTOR VESSEL TRANSPORT SYSTEM

	<u>STACKING TEST</u>
	The requirements of 49CFR 173.465(d) which requires IP-2 packages to be subjected to a stacking test for a period of at least 24 hours with a compressive load equivalent to five times the mass of the package.
49CFR 107.105(c)(2):	The specific modes of transportation for this exemption request are 1) Motor Vehicle Transportation 2) Barge Transportation The reactor vessel will be transported from the HNP facility site property by barge to the Savannah River Site (SRS) where it will then be transported by land transporter to the Barnwell facility. All transportation will be performed in accordance with the Transportation and Emergency Response Plan provided in Attachment 4.
49CFR107.105(c)(3): 49CFR107.105(c)(5):	A detailed description of the proposed exemptions follows as well as the basis for the exemption requests" The Class 7 (radioactive) materials consist of the activated reactor vessel, nozzle stubs, potentially the mirror insulation and the immovable activated reactor internals components which are grouted in place within the reactor vessel. These materials will be transported within a RVTS comprised of, (i) a Reactor Vessel Canister (hereinafter referred to as Canister) which provides the packaging, (ii) a tie-down system and, (iii) a Transportation and Emergency Response Plan. This RVTS provides safety equivalent to that of an Industrial Package Type 2 (IP-2) as described below. The HNP reactor vessel with its intact grouted reactor internals, represents "a collection of solid objects" under the definition of LSA material since each reactor internals component within the vessel and the vessel itself have concentrations of Class 7 (radioactive) materials below the LSA-III limit of 2x10-3 A2/g. On average, the reactor vessel plus the internals, excluding the

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grout, (i.e., the Class 7 material) have a specific activity of 5.26E-6 A2/g. This specific activity corresponds to about 5.3 percent of the LSA-II limit and less than 0.3 percent of the LSA-III limit. The most radioactive individual component, bottom core barrel section (41 inches) within the reactor internals, has a specific activity of 1.57E-4 A2/g. This specific activity corresponds to less than 8 percent of the LSA-III limit.

The bases for the exemption requests are due to, (i) the unique characteristics of the Class 7 (radioactive) material to be transported, (ii) its packaging and, (iii) the administrative controls that will be implemented during transportation. The basis for each exemption requested is discussed below.

<u>The shipment of the reactor vessel is one-time only, and therefore, demonstration of compliance to the regulations at the end of the exemption period is not required.</u>

<u>PACKAGING REQUIREMENT</u>

49CFR 173.427(a) requires LSA material to be packaged in accordance with paragraph (b) or (c) of this section. For LSA III material transported as an exclusive use shipment, 49CFR 173.427(b) and Table 8 of 49CFR 173.427 would require that the vessel be packaged in an Industrial Package Type 2 (IP-2). IP-2 package design and certification requirements are stipulated in 49CFR 173.411. Under the requirements of 49CFR 173.411 (b)(2), each IP-2 must meet the general design requirements of 49CFR 173.410 and prevent the loss or dispersion of radioactive material and significant increases in the radiation levels under the testing requirements of 49CFR 173.465(c) and (d) or evaluated in accordance with 49CFR 173.46 1 (a).

The applicant proposes to transport the Class 7 (radioactive) materials using an RVTS (non-specification packaging) which provides safety equivalent to an IP-2 package when transported in accordance with its Transportation and Emergency Response Plan. A description of the HNP Canister and its tie-down system is enclosed as Attachment 2. The Transportation and Emergency Response Plan is enclosed as Attachment 4.

The Canister provides containment of the Class 7 (radioactive) material by the following means:

- There will be closure of all reactor vessel penetrations.
- The reactor vessel interior will be filled with **low density** cellular concrete (LDCC) grout (25-30 lb/ft^3) to fix the surface contaminants and reactor internals components in place.
- The reactor vessel exterior surface contamination will be evaluated based on survey results. Appropriate measures will be taken to meet the LSA III requirements of 49CFR 173.
- The reactor vessel will be placed within a steel Canister with a thickness of 3 inches for containment of the activated **metals** components.
- The reactor vessel will be enclosed within the Canister with a full penetration circumferential closure weld.
- The studs, which attach the reactor vessel to the Canister, will be spray metalized to provide a seal between the Class 7 (radioactive) materials and the environs.
- The reactor vessel head, which will be attached to the Canister with studs, will be covered with a coating to fix external and internal contamination.
- The annular space between the reactor vessel and the Canister will be filled with low density cellular concrete (LDCC) grout with a nominal strength of 1,000 psi and a nominal density of 70 lb/ft^3.
- The exterior of the Canister will be painted.

This robust packaging thus includes multiple provisions to prevent release of the Class 7 (radioactive) **material** during normal transport conditions.

The RVTS provides equivalent safety to an IP-2 package by ensuring that the Canister is designed in accordance with all the general design requirements specified in 49CFR 173.410. The Canister is also designed in accordance with the additional design requirements for IP-2 packages of 49CFR 173.465(c), and (d) within the limitations of the Transportation and Emergency Response Plan.

All HNP reactor vessel activities will be controlled by the Transportation and Emergency Response Plan, which in part requires that the HNP Canister be handled in an essentially horizontal position during all transportation evolutions. Therefore, the Canister was analyzed for a 1 foot drop in the horizontal position with a 2 foot slap down at either end as opposed to the orientation, which would cause "maximum damage to the safety features being tested." Analyzing these horizontal drop scenarios is conservative for the conditions of transport regulated by the Transportation and Emergency Response Plan.

It should be noted that the transportation requirements for LSA material presented in 49CFR 173.427(a) will be met with the exception of the packaging requirements discussed above and the dose rate at 3 meters from the unshielded material specified in 49CFR 173.427(a)(1) presented below.

DOSE RATE AT 3 METERS

The unshielded reactor vessel with reactor internals within, satisfies the dose rate limitation of 10 mSv/hr as per 49CFR 173.427(a) (1). The worst case dose rate at 3 meters from the unshielded reactor vessel exterior is calculated to be 5.3 mSv/hr (530 mRem/hr). This is an estimated dose rate based on the characterization results of the Reactor Pressure Vessel and internals normalized to measured survey results obtained on the thermal shield after its removal. These normalized results were further benchmarked to surveys taken on the reactor vessel exterior on contact with the mirror insulation. The reactor vessel and the reactor internals components are considered as a collection of solid objects and it has been shown above that the worst case component concentrations are well within the limitations of 49CFR 173.403. The unshielded dose rate at 3 meters from some components, if considered separately, will exceed 10 mSv/hr. However, these internal components are an integral part of the reactor vessel.

The 3 meter radiation level requirements of Sec 173.427 (a)(1):
The basis for this requirement is loss of package shielding under normal conditions of transport and the resultant dose if the package surface radiation level exceeds 10 mSv/hr at 3 meters. The worst case dose rate at 3 meters from the unshielded reactor vessel exterior is calculated to be 5.3 mSv/hr (530 mRem/hr).

Some components with the reactor vessel, by themselves, will lead to 3 meter dose rates greater than 10 mSv/hr. However, these internals components are an integral part of the reactor vessel. They are contained inside the reactor vessel itself and surrounded by grout within the vessel. Thus, even if the integrity of the Canister is breached in its entirety under normal transport conditions, the dose rate at 3 meters from any of these components could not exceed the maximum dose rate of 5.3 mSv/hr at 3 meters from the exterior surface of the reactor vessel. Surface contamination on the activated metals was accounted for in the characterization and in determining the nuclides present.

<u>LSA III DEFINITION</u>

The definition of LSA III material includes provisions for consideration of activated metals as LSA III, but does not specifically address activated metals which are also surface contaminated. Although the definition does not include surface contaminated LSA material, the applicant does not believe it was intended to exclude activated metals with surface contamination. As a practical matter, any activated metals generated in a commercial reactor will have some level of surface contamination.

<u>LSA III LEACH TESTING REQUIREMENTS</u>

The reactor internal components and the reactor vessel interior have contaminants on their surfaces. The amount of Class 7 (radioactive) materials from surface contamination is conservatively estimated to consist of about 190 curies including about 5.2 curies of Transuranic activity. This Transuranic activity corresponds to about 149 of A2 values. These contaminants will be grouted onto their surfaces and enclosed within the reactor vessel.

The exterior surface of the reactor vessel also has surface contamination. These surface contaminants will be removed to the extent practical. Surface contamination will then be evaluated based on survey results and the appropriate measures taken to meet the LSA III requirements of 49CFR173. After placement of the reactor vessel within the Canister, LDCC will be placed in the annulus between the reactor vessel and the Canister.

We do not consider a scenario which could expose leachable surface contaminants to water for seven (7) days credible due to, (i) combination of the containment of the leachable radioactivity within the grout, (ii) the Canister design features and, (iii) the Transportation and Emergency Response Plan presented in Attachment 4.

FREE DROP TEST

Per the requirements of 49CFR 173.465(c), the package must satisfy the requirements of a drop test onto the target so as to suffer maximum damage to the safety features being tested. The drop orientation that causes ***"maximum damage"*** is typically one where the package center of gravity (cg) is located directly over one of the package comers. The package's size, weight, and handling operations constrain the package to a horizontal orientation during all transport operations once outside the Containment at the HNP. Within this framework, a 1 foot flat side drop and a 1 foot horizontal drop onto either comer followed by a 2 foot slap down represents the worst case orientation during normal conditions of transport for this package. In lieu of a physical drop test, the package was analyzed under these conditions to demonstrate compliance with the free drop requirement. A detailed discussion of the free drop test is included in Attachment 2, Transport System Description.

ATTACHMENT 1
COMPLIANCE MATRIX

EXEMPTION REQUEST FROM THE PACKAGING REQUIREMENTS OF 49CFR 173 FOR THE SHIPMENT OF THE HADDAM NECK PLANT (HNP) REACTOR VESSEL TRANSPORT SYSTEM

STACKING TEST

49CFR 173.465(d) requires IP-2 packages to be subjected to a stacking test for a period of at least 24 hours with a compressive load equivalent to five times the mass of the package. It is requested that the package be exempted from the stacking test requirement since it will be a unique one-time shipment, transported exclusive use and stacking is not credible.

49CFR 107.105(c)(4): The current project schedule requires shipment of the reactor vessel using the RVTS from the HNP facility site on or after September 1, 2000. Transport to the burial disposal facility should be accomplished within 30 to 60 days. In order to accommodate unforeseen delays in our schedule, we request that the exemption for the RVTS be applicable for two (2) years from the date of approval.

49CFR 107.105(c)(6): The Applicant is not requesting emergency processing under Sec. 107.117.

49CFR107.105(c)(7): Identification and description of hazardous material:

The estimated activity for all reactor vessel components (including reactor vessel, internals and insulation) will be approximately 809,000 curies as of September 1, 2000 (earliest shipping date of the RVTS). This estimated activity corresponds to the activity of both GTCC components (which will not be in the RVTS) and the LLRW components.

The reactor vessel package of the RVTS will only contain LLRW meeting all 10 CFR Part 61 requirements for disposal as LLRW. This LLRW includes reactor vessel internals that contain approximately 35,800 curies. When these LLRW internals are combined with the reactor vessel and insulation materials, the reactor vessel package of the RVTS will include 939,000 lbs. of activated metal containing approximately 40,500 curies. The weight of the reactor vessel package may be reduced by

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	approximately 7,600 lbs. per each reactor vessel nozzle that is not included in the reactor vessel package. (As the reactor vessel package is prepared, each of the eight reactor vessel nozzles will be removed from the outer diameter of the vessel. One or more of the reactor vessel nozzles may not be included within the components stored within the reactor vessel.) The GTCC waste (which will not be included in the reactor vessel package of the RVTS) consists of about 37,400 lbs. of activated metal and contains approximately 769,000 curies. The GTCC components are the core baffle assembly, the lower core plate, and an 89 inch section of the lower core barrel that has resided in the active fuel region. The GTCC components within the reactor vessel will be segmented and stored at the HNP site. GTCC components will <u>not</u> be included in the reactor vessel package of the RVTS. A detailed description of the characteristics of the reactor vessel and activated internal components is provided in report WMG-9913-9007, Rev. 1 entitled "Haddam Neck Reactor Vessel and Internals Characterization" and Addendum 1 thereto, which is included as Attachment 3 to this exemption request.
49CFR 107.105(c)(8):	An exemption is requested for the following shipment: The HNP reactor vessel with reactor internals approved as LSA-III material within a Canister and associated tie-down system, which is a non-specification package transported in accordance with a Transportation and Emergency Response Plan which together comprise a Reactor Vessel Transport System. A detailed description of the Canister and its tie-down system is provided in Attachment 2.
49CFR 107.105(c)(9):	CYAPCO and its contractors will perform RPV transportation activities in accordance with their respective 10CFR Part 50 Appendix B, QA programs. As **such, engineering evaluations,**

	welding and preparation of the RPV for transport will be performed in accordance with the CYAPCO QA program. CYAPCO will provide oversight of the entire project.
49CFR 107.105(d)	***Justification of exemption proposal***
49CFR 107.105(d)(1):	A description of relevant shipping and incident experience follows: The Shippingport reactor vessel was successfully shipped to Hanford, Washington under DOE regulations. The Yankee Rowe reactor vessel was successfully shipped to Barnwell, South Carolina and the Trojan reactor vessel was successfully shipped to Hanford, Washington under NRC regulations (1 0CFR 71). The Saxton reactor vessel with internals was successfully shipped to Barnwell, South Carolina under a similar DOT exemption request (DOT E-12114). Steam generators have also been successfully transported by land and water from Yankee Rowe, Salem, Trojan, Millstone, and St. Lucie.
49CFR 107.105(d)(2):	The Applicant is not aware of any increase in risk to safety or property that would result from issuing the requested exemptions.
49CFR 107.105(d)(3):	Either of the following, as applicable:
49CFR 107.105(d)(3)(i):	The applicant has designed the Canister in accordance with all the general design requirements specified in 49CFR 173.410 as well as the testing requirements 49CFR 173.465(c) and (d) within the limitations of the Transportation and Emergency Response Plan. The reactor vessel with internals is fully enclosed and grouted inside the Canister. The reactor vessel head, which has very low activity (less than 0.8 curie), is attached to and is part of the Canister. The Canister within the RVTS was evaluated to confirm the capability, in accordance with Table 12 of 173.465, **to safely** withstand a free horizontal drop of the package from a height of 1

foot onto a flat non-yielding surface without loss of containment. Horizontal drop scenarios on either end from heights of 1 foot with a slap down of 2 feet at the opposite end were considered. These evaluations are conservative relative to the limitations of the Transportation and Emergency Response Plan.

There are no other attachments or protrusions on the Canister, except structural cylindrical skirts at both ends designed to absorb energy in the event of a free drop with initial impact at the top or bottom.

Compliance with the testing requirements specified in 49CFR 173.465 for the 1 foot horizontal drop calculations is demonstrated in accordance with 49CFR 173.461(a)(4). A discussion of these calculations is provided in Section 3.2 of Attachment 2.

Due to the physical configuration of the HNP reactor vessel and the intact vessel internal components within the Canister, this exemption request does not pose increased risk to the public health and safety since there is no credible scenario under normal transport conditions resulting in direct exposure to the Class 7 material included in the reactor internals components.

A detailed discussion of the package design relative to the requirements of 49CFR 173 is provided in Attachment 2. The package design and transportation plan with the requested exemptions and alternatives achieve a level of safety equal to or greater than that of an IP-2 package.

49CFR 107.105(d)(3)(ii): This section is not applicable to this exemption request.

ATTACHMENT 2

TRANSPORT SYSTEM DESCRIPTION

HADDAM NECK PLANT

REACTOR PRESSURE VESSEL

Report WMG 9919-9007

TRANSPORT SYSTEM DESCRIPTION

HADDAM NECK PLANT

REACTOR PRESSURE VESSEL

March 2000

Report WMG 9919-9007

Prepared for:

Bechtel Power Corporation

Prepared by:

WMG, Inc.
16 Bank Street
Peekskill, NY 10566

TRANSPORT SYSTEM DESCRIPTION
HADDAM NECK PLANT
REACTOR PRESSURE VESSEL
REPORT WMG-9919-9007

WMG Inc.		16 Bank Street, Peekskill, NY 10566
Project Application	Copy No. N/A	Assigned To: N/A

APPROVALS

SIGNATURE(S) – DATE(S)

Rev No.	Preparer:	Technical Reviewer:	Approver:
0	[signature] 3/10/00	D. Shedlock 3/10/2000	[signature] 3/10/2000

FOREWORD

This report comprises Attachment 2 to the Exemption Request for the Haddam Neck Plant (HNP) Reactor Vessel Transport System. This work was performed under milestone 7 of Subcontract 24265-TSC-200.

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1.0 GENERAL INFORMATION

1.1 Introduction

This document describes the system that will be used to transport the intact Haddam Neck Plant (HNP) Reactor Pressure Vessel (RPV) containing some reactor internals components and 'potentially RPV mirror insulation from the Haddam Neck Plant (HNP) site to the Barnwell, South Carolina (S.C.) low level radioactive waste (LLRW) disposal site. This packaging is for one time use only. The radioactive materials will be transported under a DOT exemption pursuant to 49 CFR Part 107.105 via barge from the HNP site at East Hampton, Connecticut to the Savannah River Site (SRS) in Aiken, South Carolina and via land transporter from SRS to Barnwell, a distance of about 22 miles with about 0.5 miles over public highways.

The proposed Transport System consists of the "package" which is used to contain the RPV and its Class 7 (radioactive) materials, an integrated tie-down system for barge and land transport, and a Transportation and Emergency Response Plan. The exemptions requested for one-time use of this Transport System include:

1. It is requested that the package be designated as non-specification since the proposed Transport System, under normal conditions of transport and prescribed operating conditions, provides safety equivalent to that of an Industrial Package Type 2.

2. It is requested that the package be exempted from the drop requirement from the orientation that causes "maximum damage".

3. It is requested that the package contents be considered LSA III material.

4. It is requested that the package contents, including the RPV itself, be considered a "collection *of* solid objects" and that the requirement of IO mSv/hr (I R/hr) at 3 meters, as provided in 173.427(a)(1), be applied from the unshielded surface of the RPV exterior.

5. It is requested that the package contents classified as LSA-III materials be exempted from the leachability requirements of 173.468.

Section 2.0 describes the package structural design features. Packaging compliance with the requirements of 49 CFR Part 173 and the basis for compliance with these requirements is discussed in Section 3.0. The shielding analysis performed to demonstrate compliance with external radiation level requirements and the analysis results are described in Section 4.0. The basis for defining the package contents as LLRW is presented in Section 5.0. Calculation packages to support package compliance with the non-exempted requirements of 49 CFR Part 173 are referenced in this report.

1.2 Transport System Description

This section generally describes the Transport System that will be used to transport the HNP RPV with reactor internals from the HNP site to the Barnwell LLRW disposal facility.

1.2.1 Radioactive Contents

The Class 7 (radioactive) material consists of the reactor vessel, potentially the mirror insulation, and some reactor internals components and pieces thereof. All of these materials are activated stainless and carbon steel.

The radioactivity was estimated using industry accepted activation analysis methods benchmarked to radiation measurements at the RPV exterior. These analyses are documented in Attachment 3 (WMG 9913-9007, Rev. 1). As stated in this report and Addendum thereto, the metals are estimated to contain about 40,500 curies of activation product activity as of September I, 2000, the planned shipping date. During the course of plant operation, the interior surfaces of the RPV and reactor internals components were surface contaminated. The estimated radioactivity from surface contamination is about 190 curies. This amount includes about 5.2 curies of Transuranic activity. Thus, the total quantity of Class 7 (radioactive) materials is about 40,700 curies at the time of shipment.

The locations of the segmented metals components within the RPV in the shipping configuration are shown on Figure I-I. The portions of the segmented components that contain the highest curie concentration are in the vicinity of the separator plate. The distribution of activity within and including the RPV in this configuration is shown on Figure 1-2.

The reactor vessel head that will be bolted to the exterior of the canister has an estimated total activity of less than 0.8 Ci, which represents a negligible portion of the total package activity.

The total radioactivity of 40,700 curies represents about 1,910 A2 quantities with average concentrations less than 0.3 percent of the LSA III materials limit.

These contents are also LLRW, which is NRC waste form Class C Stable under the requirements of 10 CFR Part 61.

Figure 1-1

Vessel/Internals Configuration for Shipment

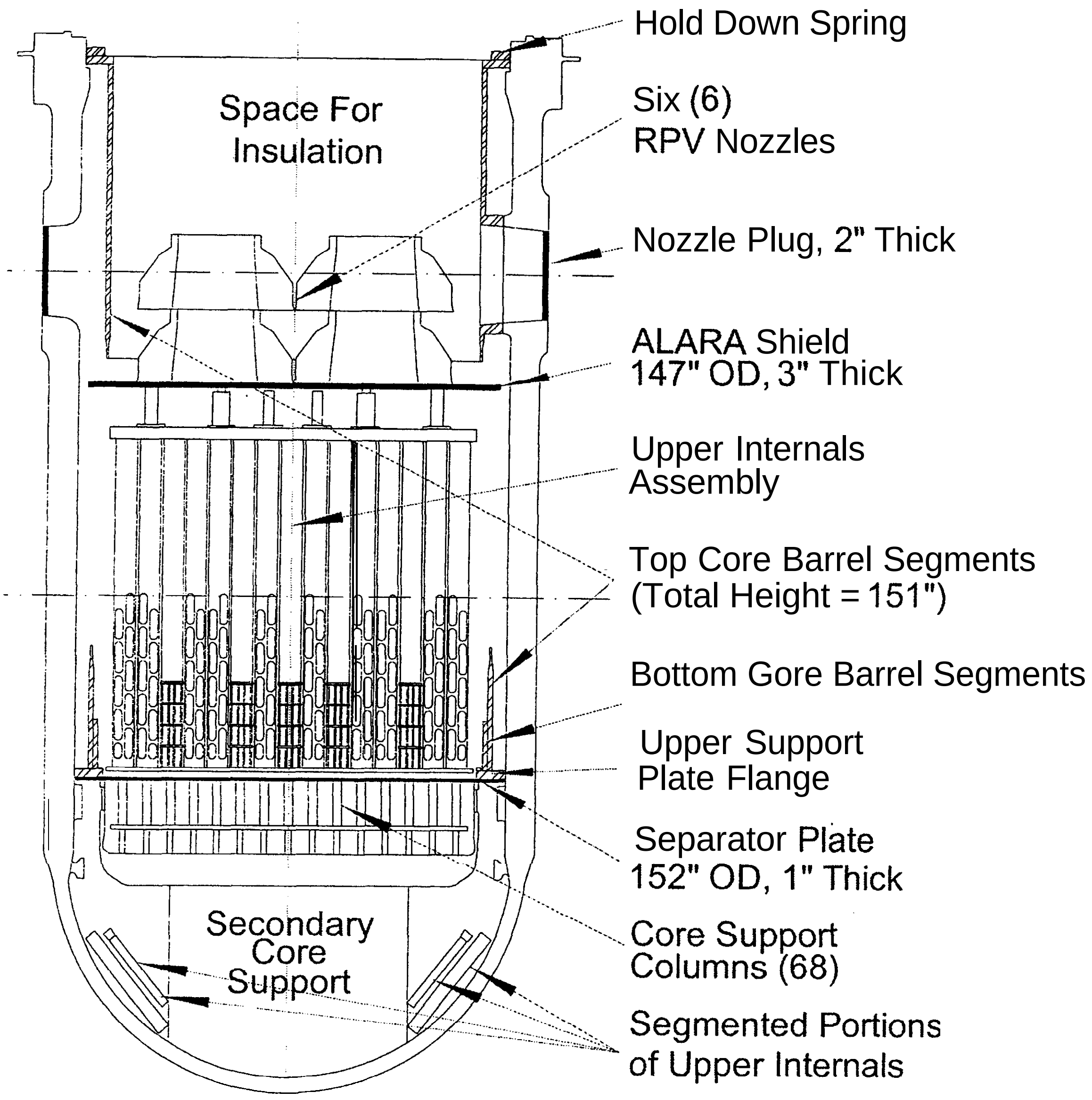
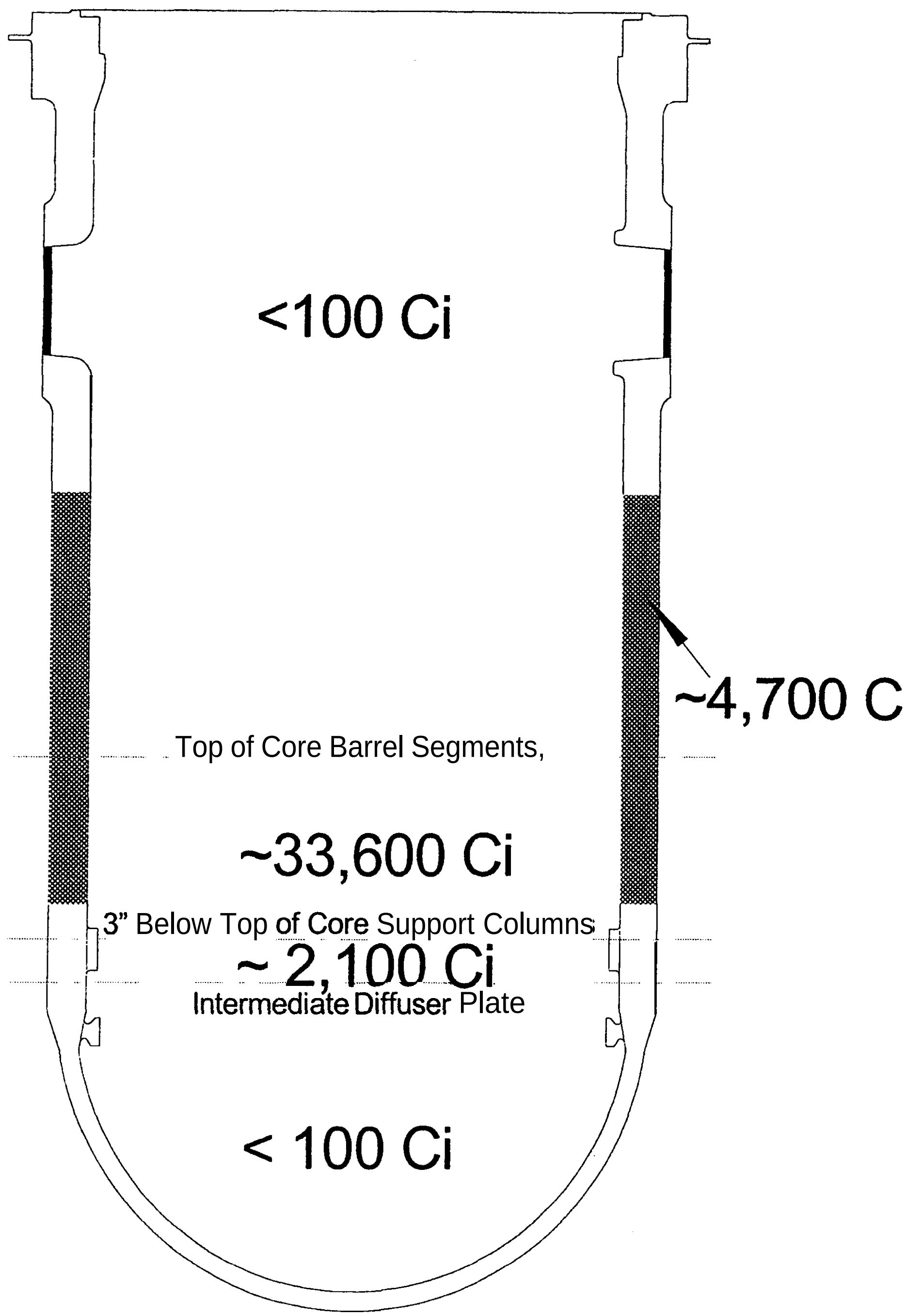


Figure 1-2

Activity Distribution Within RPV
(metals only)



1.2.2 Non-Radioactive Contents

The package contents include two un-irradiated carbon steel cylindrical flat plates located as shown on Figure l-l above. The lower (separator) plate is used as shoring to facilitate reactor internals component placement, while the upper (ALARA shield) plate is used for shielding when the RPV is handled dry. These plates have no effect on the integrity of package.

1.2.3 Package

The package is shown on Figure 1-3. It consists of a 3 inch thick carbon steel cylindrical canister with 3 inch thick top and bottom plates. As shown, an additional two inches of carbon steel is placed around its center section for shielding to meet transportation requirements. The canister comes in two sections and is closed with a full penetration field girth weld approximately 4 feet below the centerline of the RPV nozzles.

The carbon steel canister weighs about 190 tons empty and about 800 tons with its contents, including grout.

The RPV is attached to the canister by fourteen (14) RPV closure studs, which are passed through the canister top head and fastened to the upper flange of the RPV. Once the canister is fully assembled, the RPV closure stud penetrations in the canister top head are sealed with spray metalizing and the RPV head (with most, if not all, mechanical appurtenances removed) is attached by fastening to the projecting RPV closure studs. Figure 1-4 illustrates the package ready for transport. As shown, the RPV closure head is attached to the canister, and will accompany the package to the disposal facility in this configuration. The RPV head contains a negligible amount of activity, less than 0.8 curies, and represents about 72 tons of the packaging's 800 ton gross weight.

The canister is designed to provide containment comparable to that of an Industrial Package Type-2 (IP-2) under the conditions of transport bounded by this document and the other documentation submitted in support of this Exemption Request. It has been analyzed for a 1 foot horizontal drop and a 1 foot drop with 2 feet of slap down on either end.

The canister is also the component within the Transport System that provides the primary containment boundary between the RPV (with internals) components and the environs.

About 99.5 percent of the Class 7 (radioactive) material is intrinsically contained with the activated metals themselves. The remaining 0.5 percent is affixed to the metals surfaces in the form of surface contamination and is further immobilized by the grout.

Figure 1-3

Reactor Vessel Package

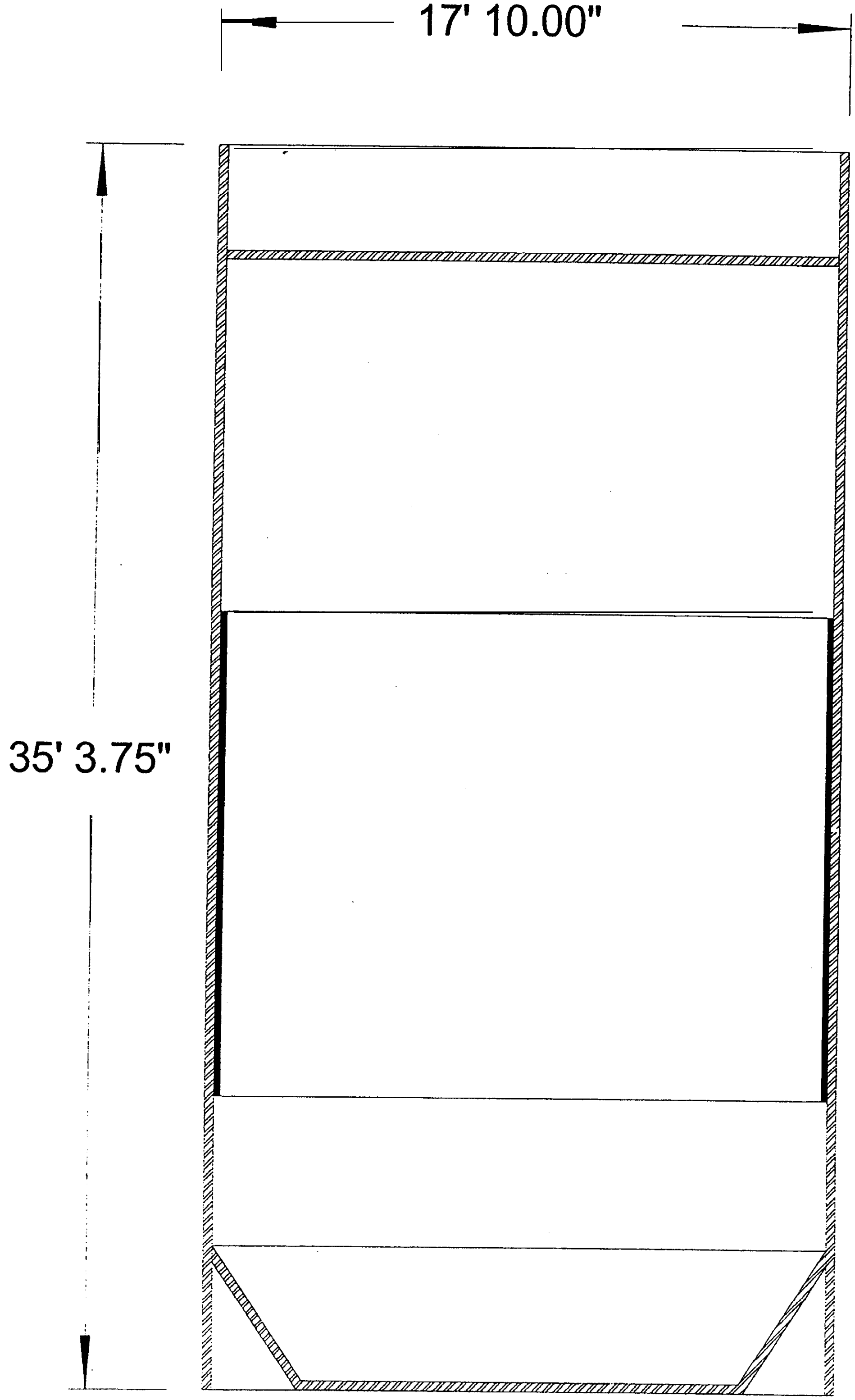
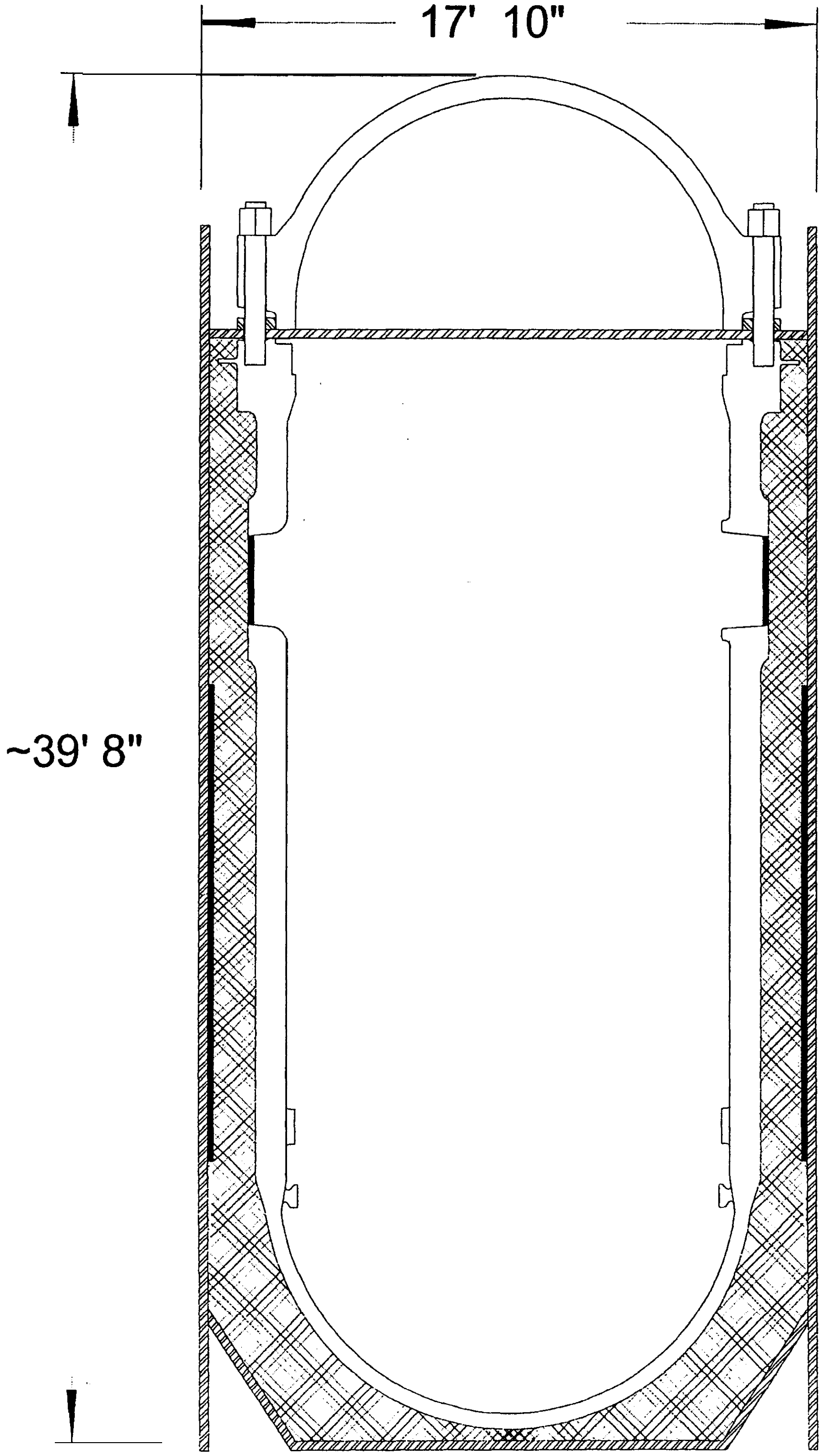


Figure 1-4

HNP Package Shipping Configuration



The major features of the package include:

1. Removal of the RPV nozzles such that they do not extend beyond the RPV upper flange,

2. Closure of all RPV bottom penetrations,

3. Closure of the eight RPV nozzles,

4. Grouting of the RPV interior to fix surface contaminants and also fix the locations of reactor internals components and pieces thereof,

5. Preparation of the RPV exterior surfaces as required to meet the . LSA III requirements of 49CFR173,

6. Grouting of the annulus between the exterior of the RPV and the interior of the canister,

7. Spray metallizing of the closure studs, which attach the RPV head to the canister at the canister top cover plate to provide a metallic seal between the radioactive contents and the environs.

These design features, which provide for multiple containment boundaries, plus the Transportation and Emergency Response Plan (Attachment 4), 'ensure that the Class 7 (radioactive) materials will be contained during transport.

1.2.4 Operational Features

Once the package is welded closed and grouted, the grout fill and vent ports will be plugged and seal welded. All temporary handling, construction aids, and lifting devices on its two sections will be removed.

There are no other valves, connections, accessible openings or other penetrations on the package.

When configured for transport as shown on Figure 1-4, the package will have no operational features.

1.2.5 Tie-down System For Transport

The package tie-down system is an integral part of the packaging design. This system, described in Section 2.5 below, consists of a skid mounted cradle provided with cables and end stops designed to secure the package during transport.

After loading the package from containment, the hydraulic land transporter will be driven onto the barge and the package will be lowered onto the barge. The package will then be attached to the barge by a system of shear keys and tie-down brackets bolted and/or welded to the barge deck.

Upon arrival at SRS, the initial loading sequence will be reversed. The package, on the tie-down system skid, will be jacked up to accept the land transporter beneath it. After lowering onto the land transporter, the skid will be attached to the land transporter by a system of shear keys and tie-down brackets welded and/or bolted to the skid and transporter frame. Transport from the SRS barge slip to the Barnwell disposal site is addressed in Attachment 4, "Transportation and Emergency Response Plan."

2.0 STRUCTURAL EVALUATION

This section describes the materials of construction and criteria used for the design and analysis of the Transport System.

2.1 Materials of Construction

The materials of construction for the package consist of:

ASTM A36 – All parts of the package and temporary attachments thereto. The package shell material and shield plates will be specified as normalized A36 or higher grade. The A36 material has a minimum yield strength of 36 ksi and a minimum ultimate strength of 58 ksi.

ASTM A572 – All structural material of the tie-down cradle and skid will be specified as ASTM A572 with a minimum yield strength of 50 ksi (Grade 50). The A572 Grade 50 material has a minimum ultimate strength of 65 ksi.

AISI-4340 (SA-193) – The original RPV closure studs will be used for the package closure. This material has a minimum yield strength of 120 ksi and a minimum ultimate strength of 135 ksi.

Carbon Steel Wire Rope - The package will be tied down with cable assemblies constructed of commercial standard IWRC (Independent Wire Rope Core) Extra Improved Plow Steel.

Spray Metalizing – Compatible metal will be used to seal the closure studs around the package top.

Weld Metals – Weld electrodes will be specified as required per American Welding Society (AWS) D1.1 – 1998, Structural Welding Code – Steel, for the applicable weld procedure and base material.

Low Density Cellular Concrete (LDCC) – The LDCC grout in the annulus between the RPV exterior and canister interior will have a nominal density of 70 lbs./ft^3. with a nominal compressive strength of 1,000 psi. The interior of the RPV will be grouted with LDCC with a density of 25 to 30 lbs./ft^3.

2.2 Package Design Criteria

The package will be designed in-accordance with the applicable requirements of the American Institute of Steel Construction (AISC) and the American Welding Society (AWS) D1.1. The design will ensure that appropriate safety factors are maintained.

2.3 Welding

The design and sizing of all package shell welds will ensure that the strength of the completed weld will be equivalent to the strength of the adjoining shell material. All welding will be performed and inspected to the requirements of AWS D1.1. Inspection of the final circumferential field closure weld may include magnetic particle testing of the root and final weld passes in addition to the required AWS inspections.

2.4 Rigging and Handling Devices

The design and operation of rigging and handling devices will be per the requirements of ASME/ANSI NQA-1, Subpart 2.15, "Quality Assurance Requirements for Hoisting, Rigging, and Transporting of Items for Nuclear Power Plants." The load bearing portions of the package required for rigging and handling will be designed in accordance with AISC requirements to ensure ample design margins are maintained.

2.5 Tie-down System

A sketch of the tie-down system is shown on Figure 2-1 and a drawing showing the principal dimensions is shown in Appendix A. As shown, the package is placed within two skid mounted cradles, which are symmetrical about the package center of gravity. Longitudinal movement is constrained by two semi-circular end stops welded to the skid, which interface with the lower section of the package skirts. Vertical restraint is provided by ten wire rope cable assemblies in basket hitch configurations, which are attached to the cradle.

2.5.1 Tie-down System Design Criteria and Analysis Results

The design of the tie-down system will be in accordance with the applicable requirements of the American Institute of Steel Construction (AISC) and the American Welding Society (AWS) D1.1 - 1998, Structural Welding Code - Steel. The system shall be subjected to loads commensurate with the most conservative loads based on 49 CFR 393 and American National Standard Institute Standards, N14.2 (Draft), N14.23, N14.24 regarding the tie-down system loads due to truck transport, shock and vibration, and maritime transport, respectively. The system design criteria are summarized in Table 2-1.

Figure 2-1

Tie-Down System

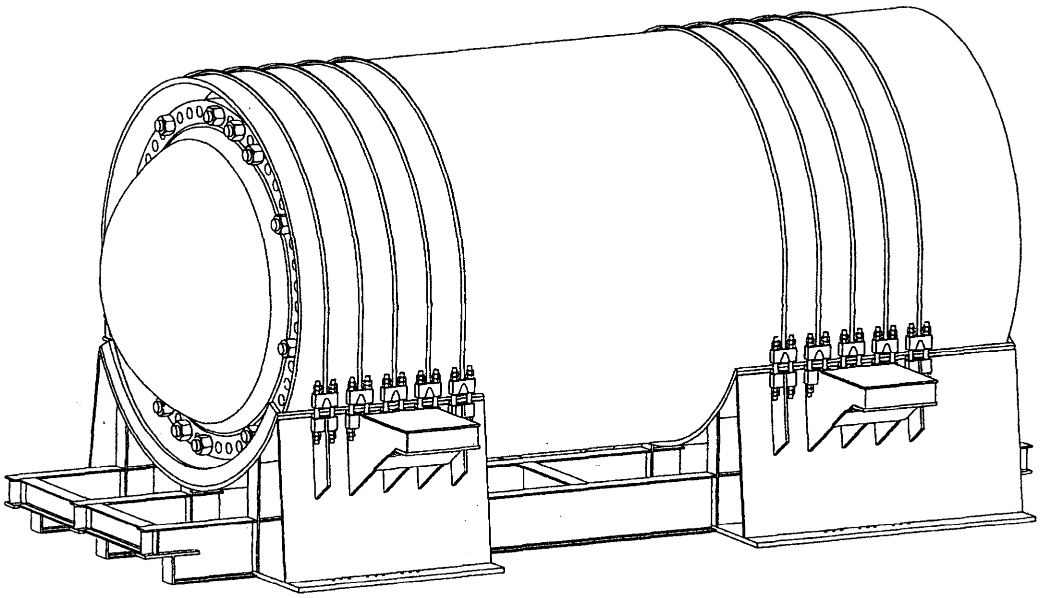


Table 2-1

Reactor Vessel Package System Tie-down Criteria

Load Orientation	DOT	Truck	Barge – Head Seas	Barge – Beam Seas
	49 CFR 393.100	**Per ANSI N14.2 (Draft Jan. 1999)**	**Per ANSI N14.24**	**Per ANSI N14.24**
Longitudinal	1.5 g	1.5 g	1.4 g	--
Vertical	1.5 g	1.5 g	2.0 g dynamic 1 .0 g static	2.0 g dynamic 1 .0 g static
Lateral	1.5 g	1.5 g	--	1.6 g
Yaw			0.1 rad/s^2	0.1 rad/s^2
Load Application	Separate	Separate	Concurrent	Concurrent

The tie-down system skid, including the wire rope cable assemblies and end stops are analyzed with standard analytical methods using the controlling loads as summarized in Table 2-1. The cradle is analyzed by finite element theory using the 3D images computer program.

The results of these analyses demonstrate that ample design margins exist relative to the applicable code allowables.

2.5.2 Attachment to Land Transporter

The tie-down system skid will be attached directly to the land transporter used to transport the package at the Savannah River site. Attachment will be by shear key and tie-down brackets bolted and/or welded to the skid and transporter frame. The transporter attachments are designed to ensure stability during transport based on the controlling load case in Table 2-1 above.

2.5.3 Attachment to Barge

The tie-down system skid will be attached to the barge. Attachment will be by shear key and tie-down brackets bolted and/or welded to the barge deck as described in Section 1.2.5. The attachments to the barge deck are designed to ensure stability during transport based on the controlling load case in Table 2-1 above.

3.0 REGULATORY COMPLIANCE DISCUSSION FOR CLASS 7 MATERIALS

This section describes the features of the package in the context of the requirements of 49 CFR Part 173 for Class 7 (radioactive) materials transported in Type 2 Industrial Packages (IP-2). Any exemptions requested relative to these requirements and the basis for the exemption is also discussed. Calculation packages, which demonstrate compliance with structural requirements, are referenced as appropriate.

3.1 General Design Requirements (173.410)

3.1 .1 Handling (173.410(a))

The package must be designed so that "The package can be *easily handled and proper/y secured.. .on a conveyance during transport."*

The package is a right circular cylinder, which will be placed horizontally on the transport conveyance within the skid mounted tie-down system described in Section 2.5 above.

3.1.2 Lifting Attachments (173.410(b))

The package must be designed so that; *"Each lifting attachment that is a structural part of the package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting attachment under excessive load will not impair the ability of the package to meet other requirements . . ."*

The package has no lifting devices that will be used offsite during transportation.

The package must be designed so that; *"Any* other structural part of the *package which* could *be used to lift the* package must *be capable of being rendered inoperable for lifting the package during transport or ..."*

There are no lifting devices on the package and no structural part of the package can be used for lifting as shown on Figure 1-4. All temporary lifting devices used during packaging operations will be removed or disabled at the HNP site prior to shipment.

3.1.3 Exterior Protrusions (173.410(c))

The package must be designed so that; *"The external surface, as far as practicable, will be free from protruding features and will be easily decontaminated."*

There are no protruding features on the exterior of the package (see Figure 1-4). External surface preparation will ensure ease of decontamination.

3.1.4 Water Collection Pockets (173.410(d))

The package must be designed so that; *"The outer layer of packaging will avoid, as far 'as practicable, pockets or crevices where water might collect."*

Both the top and bottom skirts, which may have the potential to collect rainwater, will have drain holes at their lowest points to eliminate such water collection.

3.1.5 Feature Safety Impacts (173.410(e))

The package must be designed so that; *"Each feature that is added to the package will not reduce the safety of the package."*

No features have been added to the package which reduce its safety.

3.1.6 Normal Transport Vibrations (173.410(f))

The package must be designed so that; *"The package will be capable of withstanding the effects any acceleration, vibration or vibration resonance that may arise under normal conditions of transport without any deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole and without loosening or unintentionally releasing the nuts, bolts, or other securing devices even after repeated use."*

The package will be transported via barge and land transporter and has been designed to withstand the vibration requirements defined in ANSI 14.23. The package was analyzed for vibrations and these analyses are presented in Reference 14. The results of these analyses indicate that no degradation of the effectiveness of the package will occur due to the vibrational loads.

3.1.7 Chemical Compatibility (173.410(g))

The package must be designed so that; *"The materials of construction of the packaging and any components or structure will be physically and chemically compatible with each other and the package contents.. ."*

The materials used for the package include carbon steel plate, welding electrodes, stainless steel (studs), carbon steel from spray metalizing, low density cellular concrete, and a coating system on the RPV and package exteriors. All these materials are compatible with each other.

3.1.8 Valves (173.41 0(h))

The package must be designed so that; *"All valves through which the package contents could escape will be protected against unauthorized operation."*

There are no valves on the package.

3.2 Free Drop Under 173.465(c) As Per (1 73.411 (b)

Under 173.465(c), IP-2 packages must satisfy the requirements for the free drop test, which provides: *"The specimen must drop onto the target so as to suffer maximum damage to the safety features being tested."*

It is requested that the package be exempted from the drop requirement from the orientation that causes *"maximum damage".* Such a drop orientation is typically one where the package cg is located directly over one of the package corners. The package's size, weight, handling operations, and the Transportation and Emergency Response Plan (see Attachment 4) constrain the package to a horizontal orientation during all transport operations once outside the Containment at the HNP site.

Within this framework, a 1 foot flat side drop and a 1 foot horizontal drop onto either corner followed by a 2 foot slap down represents the worst case orientation during normal conditions of transport for this package. The package was analyzed under these conditions to demonstrate compliance with the free drop requirement.

The analysis was performed using classical energy balance methods for impact loads using conservative assumptions to determine crush strength and maximum deformations resulting from the postulated worst case drop scenarios.

3.2.1 Analysis Results

3.2.1.1 Loss or Dispersion of Contents

The results of the impact analyses indicate that the integrity of the package and welds are not compromised by the deformation resulting from the impact loads due to the postulated drop conditions. Moreover, the intrinsic nature of the package contents, grouted solid objects, prevents dispersion.

3.2.1.2 Increase in Radiation Levels

The internals components are grouted in position within the RPV, and under worst case conditions contribute less than 10 percent to the external dose rates (see Section 4.0 below). The internals pieces with the highest specific activities include the pieces of the lower core barrel placed around the periphery, and the core support columns (see Figure l-l). These pieces are not expected to shift significantly during the analyzed horizontal free drop.

Additionally, the shielding from the package shell and the supplemental shielding in the core region remain intact during the free drop.

Accordingly, no increase in external radiation levels is expected to result from the free drop effects.

3.3 Stacking Test 173.465(d) As Per (173.411 (b))

Under 173.465(d), IP-2 packages must be; *"subjected for a period of at least 24 hours to a compressive load equivalent to.. .five times the mass of the package.. ."*

We request that the package be exempted from the stacking requirement since it will be a one-time shipment, transported exclusive use and the stacking load is not credible.

3.4 Contamination Controls (173.427(a)(4))

Packages must meet the contamination control limits specified in 173.443.

The package will be new and used for a single shipment. When fabricated, the exterior will be coated with paint. The procedures employed for contamination control prior to shipment will conform to those used for all radioactive material shipments from the HNP site.

3.5 Thermal Limitations (173.442)

"A package of Class 7 (radioactive) material must be designed, constructed, and loaded so that - (a) The heat generated within the package by the radioactive contents will not, during conditions normally incident to transport, effect the integrity of the package, and.. . "

The heat load from package contents including the RPV itself is calculated at 280 watts with the major contributor being the 18,800 curies of Co-60. A simplified conduction model was developed to transmit this heat load from the exterior of the RPV to the exterior surface of the package. This model conservatively assumed that the entire heat load was in the IO foot high core region of the RPV wall. The results indicated that this heat load will increase the surface temperature of the package by less than 1 degree F. This temperature increase will have no effect on the integrity of the package.

"(b) The temperature of the accessible external surfaces of the loaded package will not, assuming still **air in** *the* **shade at an ambient temperature of** *38 degrees C (100 degrees F), exceed.. .(2) 85 degrees C (185 degrees F) in an exclusive use shipment. "*

The internal heat load will have a negligible effect on the ambient temperature of the package external surface.

3.6 Low Specific Activity (LSA) Material Requirements

3.6.1 Definitions

LSA III materials are *"solids (e.g., consolidated wastes, activated materials) that meet the requirements of 173.468 and which:*

(i) The Class 7 (radioactive) material is distributed throughout a solid or collection of solid objects, or is essentially . .."

The radioactive materials consist of activated materials in the form of the RPV, reactor internals components and pieces thereof and potentially the mirror insulation which surrounded the exterior of the RPV. Each of these activated materials is also coated with surface contamination in the form of activation products, fission products, and Transuranics. The total radioactivity within the package is about 40,700 curies consisting of about 40,500 curies from activation plus about 190 curies from surface contaminants. Of the total attributed to surface contamination there are about 5.2 curies of Transuranics.

Two exemptions are requested;

1. That the package contents be considered LSA III material, even though surfaces are contaminated. The basis for this request is that all activated metals will have relatively small amounts, in this case less than 0.5 percent, of surface contaminants. Since almost all of the radioactivity arises from activation, we request that the contents be considered LSA III materials.

2. That the package contents, including the RPV itself, be considered a *"collection of solid objects"* and that the requirement of IO mSv/hr (I R/hr) at 3 meters, as provided in 173.427(a)(1), be applied from the unshielded surface of the RPV exterior. The basis for this request is that while some of the individual components or pieces thereof within the RPV will exceed 1 0mSv/hr (I R/hr) at 3 meters, they are themselves contained and fixed within the RPV. Therefore, under any scenario that could lead to breach of the package integrity, the only source of radiation exposure would be that from the RPV itself and the dose rates would be less than the 1R at 3m.

3.6.2 LSA III Material Leachability (173.468)

LSA III materials are *"solids (e.g., consolida ted wastes, activated materials) that meet the requirements of 1 73.468 and which:*

(ii) The Class 7 (radioactive) material is relatively insoluble, or is intrinsically contained in a relatively insoluble ma terial, so that, even under loss of packaging, the loss of Class 7 (radioactive) material per package by leaching when placed in water for 7 days would not exceed 0.1 A2"

An exemption from this requirement is requested because;

I. The bulk of the radioactivity arises from activation and is intrinsically contained within a relatively insoluble material consisting of carbon and stainless steel, and

2. That portion of the radioactivity not intrinsically contained arises from surface contamination in the amount of about 190 curies on the interior surface of the RPV and internals components. These radioactive materials will be within the RPV and grouted.

3. The remaining surface contamination not intrinsically contained will be on the exterior of the RPV. This surface will be prepared to meet the LSA Ill requirements of 49CFR173 before placement within the package and then grouted.

Accordingly, we do not consider a scenario which could expose leachable surface contaminants to water for seven days credible due to the containment of the leachable radioactivity, the package design features and the Transportation and Emergency Response Plan presented in Attachment 4.

3.6.3 LSA Limit Calculations

LSA III materials are *"solids [e.g., consolidated wastes, activated materials) that meet the requirements of 173.468 and which;*

(iii) the average specific activity of the solid does not exceed 2e-3 A2/g."

The package contents on the average, and for the worst case component, have A2/g concentrations well below the LSA III limit. The analytical results to support these conclusions are presented below.

3.6.3.1 Average LSA Concentrations

Table 3-1 presents the DOT classification summary for the Class 7 (radioactive) materials within the package. For a radioactive materials weight of 800,000 lbs. the A2/g value is 5.26E-06. This is less than 0.3 percent of the LSA Ill limit and only about 5.3 percent of the LSA II limit.

As also shown on this exhibit, the package contains 1,910 A2 quantities of radioactive materials with about 149 A2 quantities consisting of Transuranic surface contaminants.

Table 3-1

HNP Reactor Vessel and Internals
DOT Classification Summary

Radionuclide	Neutron Activation Curies	Surface Contamination Curies	Total Activity Curies	A2 Value Curies	Type A Fraction	LSA A2/g Value
H-3	NP	LLD	LLD	1.08E+03	0.00E+00	0.00E+00
C-14	3.75E+00	LLD	3.75E+00	5.41E+01	6.93E-02	1.91E-10
Mn-54	2.77E+02	9.11E-01	2.78E+02	2.70E+01	1.03E+01	2.83E-08
Fe-55	1.87E+04	3.89E+01	1.87E+04	1.08E+03	1.74E+01	4.78E-08
Co-57	NP	9.36E-02	9.36E-02	2.16E+02	4.34E-04	1.19E-12
Co-60	1.87E+04	1.37E+02	1.88E+04	1.08E+01	1.74E+03	4.80E-06
Ni-59	2.12E+01	LLD	2.12E+01	1.08E+03	1.96E-02	5.41E-11
Ni-63	2.83E+03	8.61E+00	2.84E+03	8.11E+02	3.50E+00	9.65E-09
Sr-90	NP	2.20E-02	2.20E-02	2.70E+00	8.13E-03	2.24E-11
Nb-94	5.90E-02	LLD	5.90E-02	1.62E+01	3.64E-03	1.00E-11
Tc-99	1.91E-02	LLD	1.91E-02	2.43E+01	7.85E-04	2.16E-12
I-129	NP	LLD	LLD	Unlimited	0.00E+00	0.00E+00
Cs-137	NP	LLD	LLD	1.35E+01	0.00E+00	0.00E+00
Np-237	NP	6.02E-04	6.02E-04	5.41E-03	1.11E-01	3.07E-10
Pu-238	NP	2.54E-01	2.54E-01	5.41E-03	4.69E+01	1.29E-07
Pu-239/240	NP	8.12E-02	8.12E-02	5.41E-03	1.50E+01	4.13E-08
Pu-241	NP	4.54E+00	4.54E+00	2.70E-01	1.68E+01	4.63E-08
Am-241	NP	2.48E-01	2.48E-01	5.41E-03	4.59E+01	1.26E-07
Cm-242	NP	1.53E-03	1.53E-03	2.70E-01	5.66E-03	1.56E-11
Cm-243/244	NP	8.99E-02	8.99E-02	8.11 E-03	1.11E+01	3.05E-08
Totals	4.05E+04	1.91E+02	4.07E+04		1.91E+03	5.26E-06

Waste Weight 800,000 lb
3.63E+08 g

LSA II Limit 1.00E-04
Percent LSA II Limit 5.26%

Definitions:

NP = Not Present
LLD = Lower Limit of Detection

3.6.3.2 Worst Case Bottom Core Barrel Section (41 inches)

The component within the package with the highest concentration of activation products is the bottom core barrel section (41 inches). This component is a segment of the core barrel (see Attachment 3 Section 2.2.2 and Addendum 1). The bottom core barrel section is estimated to contain about 13,700 curies and weigh about 9,480 lbs.

Table 3-2 presents the DOT classification summary for the worst case material within the package. For a radioactive materials weight of 9,480 lbs. the A2/g value is 1.57E-4. This value is less than 8 percent of the LSA III limit.

As also shown on this exhibit, this component contains 676 A2 quantities of radioactive material with less than 3 A2 of Transuranic surface contaminants.

Table 3-2

HNP Bottom Core Barrel Section (41 inches)
DOT Classification Summary

Radionuclide	Neutron Activation Curies	Surface Contamination Curies	Total Activity Curies	A2 Value Curies	Type A Fraction	LSA A2/g Value
H-3	NP	LLD	LLD	1.08E+03	0.00E+00	0.00E+00
C-14	1.36E+00	LLD	1.36E+00	5.41 E+01	2.51E-02	5.83E-09
Mn-54	7.81E+01	1.96E-02	7.81 E+01	2.70E+01	2.89E+00	6.72E-07
Fe-55	5.43E+03	8.36E-01	5.43E+03	1.08E+03	5.03E+00	1.17E-06
Co-57	NP	2.01E-03	2.01 E-03	2.16E+02	9.32E-06	2.17E-12
Co-60	7.17E+03	2.95E+00	7.17E+03	1.08E+01	6.64E+02	1.54E-04
Ni-59	7.73E+00	LLD	7.73E+00	1.08E+03	7.15E-03	1.66E-09
Ni-63	1.03E+03	1.85E-01	1.04E+03	8.11 E+02	1.28E+00	2.97E-07
Sr-90	NP	4.72E-04	4.72E-04	2.70E+00	1.75E-04	4.07E-11
Nb-94	2.31E-02	LLD	2.31 E-02	1.62E+01	1.42E-03	3.31E-10
Tc-99	5.13E-03	LLD	5.13E-03	2.43E+01	2.11E-04	4.91E-11
I-129	NP	LLD	LLD	Unlimited	0.00E+00	0.00E+00
Cs-137	NP	LLD	LLD	1.35E+01	0.00E+00	0.00E+00
Np-237	NP	1.29E-05	1.29E-05	5.41E-03	2.39E-03	5.56E-10
Pu-238	NP	5.45E-03	5.45E-03	5.41 E-03	1.01E+00	2.34E-07
Pu-239/240	NP	1.74E-03	1.74E-03	5.41 E-03	3.23E-01	7.50E-08
Pu-241	NP	9.75E-02	9.75E-02	2.70E-01	3.61E-01	8.40E-08
Am-241	NP	5.33E-03	5.33E-03	5.41 E-03	9.86E-01	2.29E-07
Cm-242	NP	3.28E-05	3.28E-05	2.70E-01	1.22E-04	2.83E-11
Cm-243/244	NP	1.93E-03	1.93E-03	8.11 E-03	2.38E-01	5.54E-08
Totals	1.37E+04	4.11E+00	1.37E+04		6.76E+02	1.57E-04

Waste Weight 9.48E+03lb
4.30E+06g

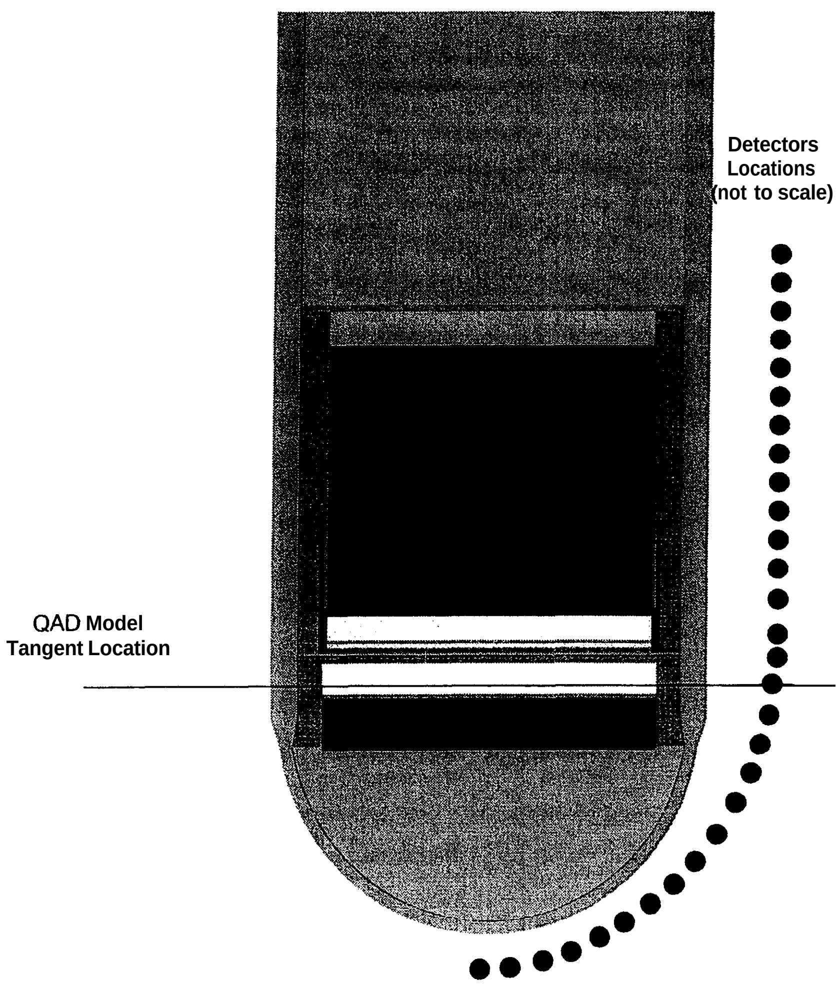
LSA III Limit 2.00E-03
Percent LSA III Limit 7.86%

Definitions:
NP = Not Present
LLD = Lower Limit of Detection

4.0 SHIELDING EVALUATION

The shielding evaluation was performed using QAD-CGGP-A (QAD). QAD is a point kernel shielding program that uses Combinatorial Geometry (CG) and Geometric Progression (GP) buildup factors for performing neutron and gamma ray shielding calculations. The combinatorial geometry allows the construction of complex three-dimensional models. PICTURE, a program for generating two-dimensional, color cross sections of the CG, was used to verify the accuracy of the geometry. Figure 4-1 is a Y-Z cross section of the HNP RPV and internals, with the internals in their anticipated shipping configuration. The RPV head is not included in the shielding model. The bulk of the activity is contained in the lower portions of the vessel and a significant amount of steel shielding is provided by the ALARA shield (3"), closure plate (3"), and the vessel head itself (8") for a total of 14" of steel shielding. The significant amount of steel shielding, 14" relative to the 10.75" thick vessel wall, coupled with the increased distance from the source region, clearly indicates that the exposure rates through the vessel wall are bounding.

Figure 4-1
PICTURE Code Generated
Y-Z Cross Section of the HNP RPV and Internals



4.1 Source Term Definition

4.1 .1 Radioactive Content

Because of the relatively long cooling time, it is reasonable to assume that the measured dose rate will be from Co-60. The Co-60 source terms used for this analysis were those calculated by WMG in activation analysis report, WMG-9913-9007, Rev. 1 with activities as of September 1, 2000 to . support the planned shipping date. The reactor vessel and internals models considered the dose contribution for the following 11 source regions:

1. Reactor Vessel Wall – carbon steel base metal
2. Vessel Cladding – stainless steel cladding on the interior of the reactor vessel wall
3. Core Barrel Bottom Segmented Section – 41 inch piece of the lower core barrel remaining after segmentation
4. Core Barrel Top Segmented Section – 151 inch piece of the upper core barrel remaining after segmentation
5. Intermediate Diffuser Plate – diffuser plate located between the lower core support plate and the lower core plate
6. Support Column (2) – cylindrical section of the support columns directly on top the intermediate diffuser plate
7. Support Column (1) – section of the support columns that flare outward, just below the lower core plate
8. Upper Core Plate – stainless steel plate for holding the fuel in position
9. Orifice Plate Region – circular plates on top of the upper core plate that attach the support columns to the upper core plate.
10. Upper Support (1) – 24 inch section of the upper support columns on top of the orifice plate region
11. Upper Support (2) – Remaining section of the upper support columns that extends up to the deep beam region

Table 4-1 is a summary of the different components that were defined as source terms with their Co-60 curie content.

Table 4-1

Co-60 Curie Content of Source Regions

Source Regions	Co-60 Activity as of (9/1/2000) Activity (Ci)	(dps)
Vessel Wall	3.75E+02	1.39E+13
Vessel Clad	2.25E+02	8.32E+12
Core Barrel Bottom	6.59E+03	2.44E+14
Core Barrel Top	7.57E+03	2.80E+14
Intermediate Diffuser Plate	2.48E+01	9.18E+11
Support Column (2)	7.95E+02	2.94E+13
Support Column (1)	1.67E+03	6.18E+13
Upper Core Plate	6.54E+02	2.42E+13
Orifice Plate Region	1.03E+02	3.81E+12
Upper Support (1)	6.39E+01	2.36E+12
Upper Support (2)	3.06E+00	1.13E+11

4.1.2 Shielding Description

The shielding for the RPV and internals consists of a 3 inch carbon steel canister with additional steel shielding adjacent to, and extending 2 feet above and below the active fuel region. Seventy (70) lb/ft^3 (1,000 psi) grout fills the void between the RPV and the canister. The internals are also grouted in place using 25-30 lbs/ft^3 low density cellular concrete (LDCC) grout. However, for the shielding analysis the additional shielding from grout was neglected to ensure conservatism in the shielding design.

4.2 Three (3) Meter Dose Rate from Unshielded Source

This section describes the QAD models used to quantify the dose rate at 3 meters from the unshielded surface of the RPV with the mirror insulation removed. The internals are positioned in their anticipated shipping configuration as shown in Figure 4-1. No internal grout was considered in the model.

4.2.1 Analytical Model

A master combinatorial geometry for the reactor vessel and the internals was used for all model inputs. Separate models were run with the source defined in each region to determine the contribution to the measured dose rate at each detector location. The results from each region were then summed to get the total dose rate at each detector location.

All detectors were located 3 meters from the RPV. Starting directly below the RPV bottom head, detectors are placed at 1 foot arc lengths up to the RPV tangent location (see Figure 4-1). From the tangent location, detectors were placed axially in 1 foot increments to a height of about 2 feet above the active fuel. The wide spread of detector locations demonstrates that the maximum dose is in the locale of core centerline.

4.2.2 Analysis Results

The analysis results indicate the maximum dose rate at 3 meters is about 530 mR/hr. The locale of the maximum dose rate is about 7 feet above the RPV wall transition near the axial core centerline. Figure 4-2 displays the variation in dose rate as a function of height from the RPV wall tangent location. Because of the RPV thickness, 99 percent of the external dose at 3 meters is from the RPV wall. Below the transition region most of the measured activity is from the internals. This is because about 90 percent of the RPV activity is located adjacent to the active fuel. The remaining 10 percent is located ± 2 feet of the active fuel. Table 4-2 summarizes the 3 meter dose rates for the HNP RPV and internals. The 180 degree location is directly below the RPV bottom head. 90 degrees would coincide with the 0 ft detector location, the RPV wall tangent location (see Figure 4-1).

Figure 4-2

3 Meter Dose Rate (mR/hr)

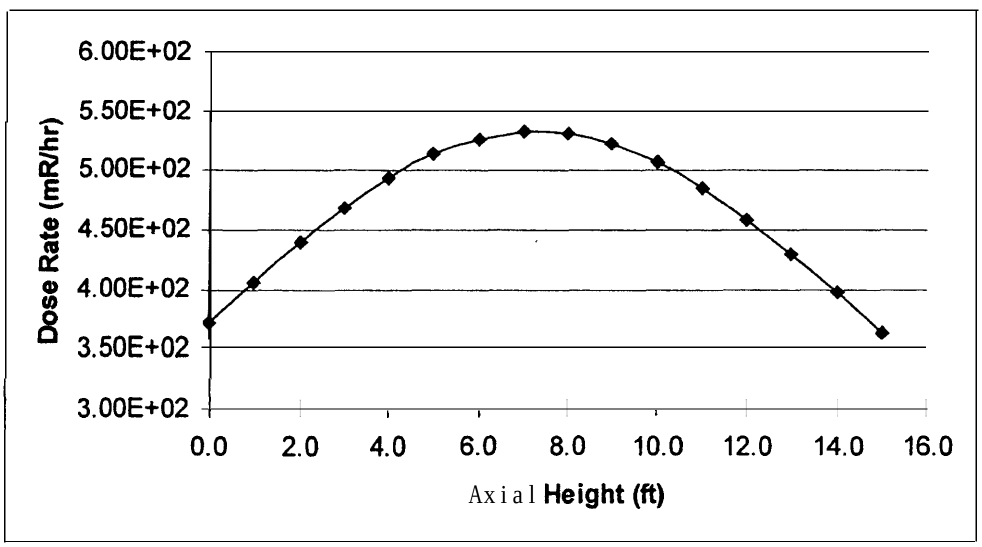


Table 4-2
3 Meter Dose Rates at Different External Locales
As of September 1, 2000

Detector Location		Dose Rate (mR/hr)	Component Fractional Dose Contribution				
			Wall	Clad	Core Barrel Bottom Segment	Core Barrel Top Segment	Remaining
180.0	degrees	1.90E-02	0.00	0.01	0.03	0.02	0.94
178.0	degrees	1.91E-02	0.00	0.01	0.03	0.02	0.94
171.3	degrees	2.11E-02	0.02	0.07	0.02	0.02	0.86
164.5	degrees	3.04E-02	0.04	0.31	0.01	0.02	0.62
157.7	degrees	2.60E-01	0.00	0.03	0.06	0.76	0.15
150.9	degrees	8.08E+00	0.91	0.00	0.01	0.05	0.03
144.2	degrees	3.46E+01	0.96	0.00	0.00	0.01	0.02
137.4	degrees	6.58E+01	0.96	0.00	0.00	0.00	0.04
130.6	degrees	9.93E+01	0.95	0.00	0.00	0.00	0.04
123.9	degrees	1.35E+02	0.95	0.00	0.00	0.00	0.05
117.1	degrees	1.71E+02	0.97	0.00	0.00	0.00	0.03
110.3	degrees	2.10E+02	0.99	0.00	0.00	0.00	0.01
103.5	degrees	2.57E+02	0.99	0.00	0.00	0.00	0.00
96.8	degrees	3.13E+02	0.99	0.00	0.00	0.01	0.00
0.0	ft	3.72E+02	0.98	0.00	0.00	0.01	0.01
1.0	ft	4.07E+02	0.98	0.00	0.00	0.01	0.01
2.0	ft	4.40E+02	0.98	0.00	0.00	0.01	0.01
3.0	ft	4.69E+02	0.98	0.00	0.00	0.01	0.00
4.0	ft	4.94E+02	0.98	0.00	0.00	0.01	0.00
5.0	ft	5.13E+02	0.99	0.00	0.00	0.01	0.00
6.0	ft	5.27E+02	0.99	0.00	0.00	0.01	0.00
7.0	ft	5.33E+02	0.99	0.00	0.00	0.01	0.00
8.0	ft	5.31E+02	0.99	0.00	0.00	0.00	0.00
9.0	ft	5.22E+02	1.00	0.00	0.00	0.00	0.00
10.0	ft	5.07E+02	1.00	0.00	0.00	0.00	0.00
11.0	ft	4.85E+02	1.00	0.00	0.00	0.00	0.00
12.0	ft	4.59E+02	1.00	0.00	0.00	0.00	0.00
13.0	ft	4.29E+02	1.00	0.00	0.00	0.00	0.00
14.0	ft	3.97E+02	1.00	0.00	0.00	0.00	0.00
15.0	ft	3.65E+02	1.00	0.00	0.00	0.00	0.00

4.3 Dose Rates from Package Exterior during Transport (173.441)

This section describes the QAD models used to quantify the dose rates from the reactor vessel package exterior during transportation. The internals are positioned in their anticipated shipping configuration as shown in Figure 4-1. Source terms considered for the calculation are discussed in Section 4.1. To ensure conservatism in the estimated package dose rates, no internal grout was used in the shielding models. However, cases were run with and without external grout in the annulus between the RPV exterior and the canister to determine the effects of the grout on external dose rates. Dose rates were determined on contact, at 1 foot and at 2 meters from the package. All dose rates are based on the anticipated shipping date of September 1, 2000.

4.3.1 Shielding Configuration

The package shielding configuration is shown in Figure 4-3. The canister itself is 3 inches thick. The additional shielding on the interior of the canister was modeled over a range of thicknesses to determine the amount of steel shielding required to meet transportation dose requirements as follows:

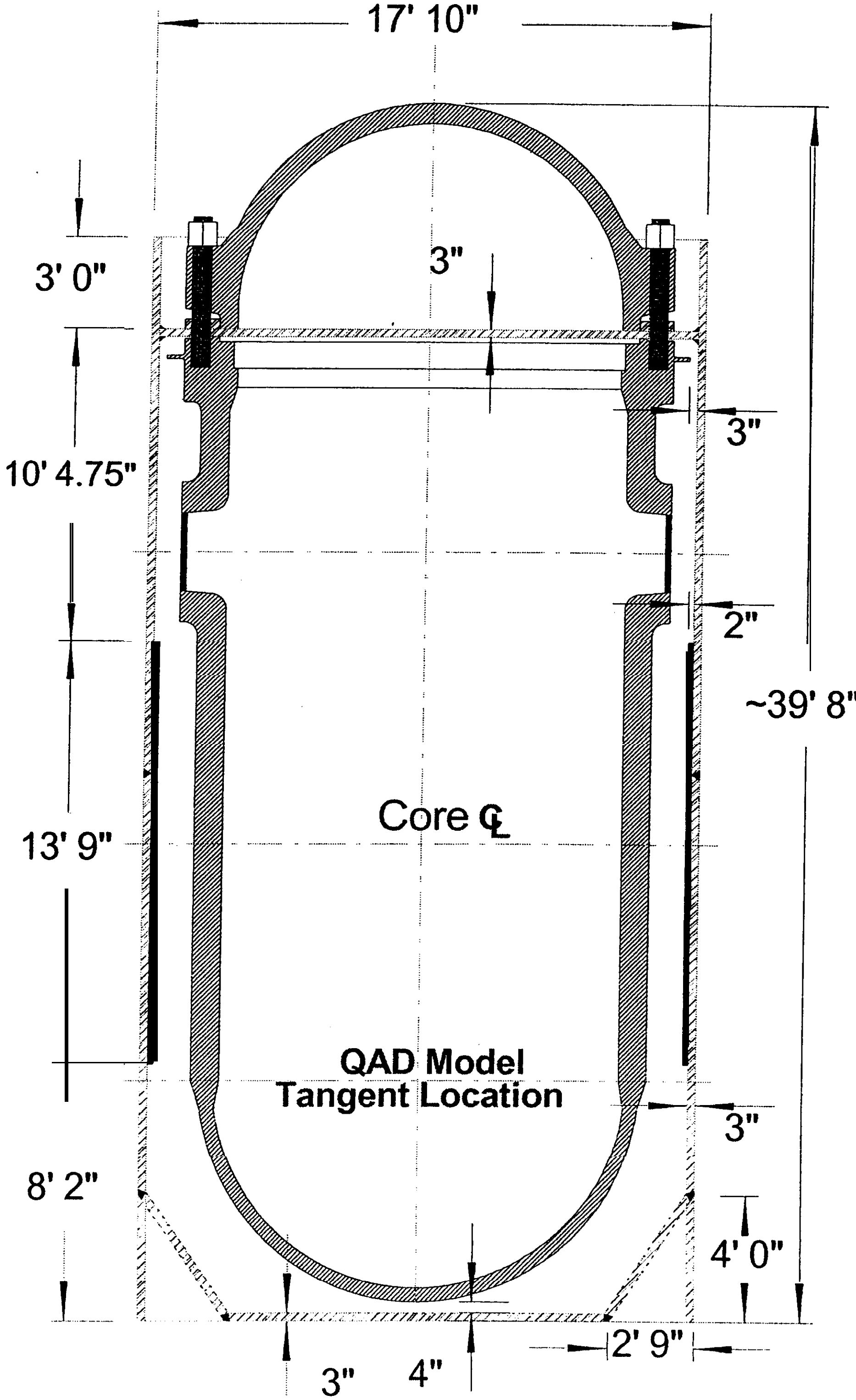
1. Less than 200 mRem/hr on the external surface of the package,
2. Less than 200 mRem/hr at any point on the vertical planes projected from the outer edges of vehicle, on the upper surface of the load, and on the lower external surface of vehicle,
3. Less than 10 mRem/hr at any point 2 meters from the vertical planes projected from the side of the transport conveyance,
4. Less than 2 mRem/hr in any normally occupied spaces.

The additional shielding is located on the interior of the disposal container, adjacent to the active fuel region, extending about 2 feet above and 2 feet below this region. Additional shielding is only required in this region for two reasons:

1. More than 99 percent of the RPV activity is located within ± 2 feet of the active fuel region.
2. More than 99 percent of the external dose comes from the RPV wall.

Figure 4-3

HNP Reactor Vessel Package Shielding Configuration



4.3.2 Analytical Model

A master combinatorial geometry for the reactor vessel and the internals was used for all model inputs. Separate models were run with the source considered in each region to determine the contribution to the measured dose rate at each detector location. The results from each region were then summed to get the total dose rate at each detector location.

Detectors were placed axially in 1 foot increments, starting from the vessel wall tangent locale (see Figure 4-1) until a height of about 2 feet above the active fuel region was reached. Detectors were radially located on contact (1 inch), at one foot and at 2 meters.

4.3.3 Analysis Results

The analysis results indicate that the 5 inches of steel shielding comprised of the 3 inch canister wall plus 2 inches of shielding in the fuel region are adequate shielding to meet dose rate requirements. The external grout between the reactor vessel and the reactor vessel canister has a nominal density of about 70 lbs/ft^3 (1,000 psi grout). The analysis was conservatively run assuming the 9 inch annulus space had a grout density of 50 lbs/ft^3 grout. These results indicated dose rates of about 3.7 mR/hr on contact and 1.7 mR/hr at 2 meters from the package surface. It is not expected that measured dose rates will exceed these values.

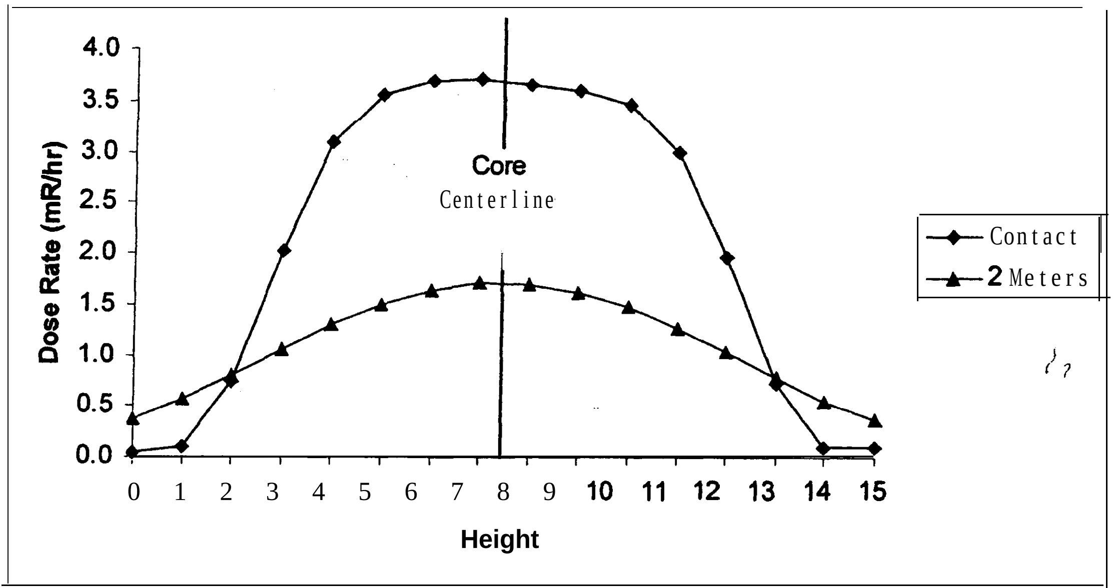
Table 4-3 summarizes the results of the reactor vessel package external dose rate analysis. Figure 4-4 demonstrates the variation in axial dose rates on the reactor vessel package.

Table 4-3
Reactor Vessel Package External Dose Rates (mR/hr)
As of September 1, 2000

Height From Vessel Tangent (ft)	5" Shield with Annulus Grout at 50 lbs/ft^3	
	Contact (mR/hr)	2 meter (mR/hr)
0	0.0	0.4
1	0.1	0.6
2	0.7	0.8
3	2.0	1.1
4	3.1	1.3
5	3.6	1.5
6	3.7	1.6
7	3.7	1.7
8	3.6	1.7
9	3.6	1.6
10	3.4	1.5
11	3.0	1.3
12	2.0	1.0
13	0.7	0.8
14	0.1	0.5
15	0.1	0.4

Figure 4-4

External Dose Rate With Annulus Grout (mR/hr)



5.0 WASTE CLASSIFICATION UNDER 10 CFR PART 61

The contents of the package are low level radioactive waste and were classified according to the requirements of 10 CFR Part 61. The waste form class of each component that comprises the contents of the package is presented in Attachment 3, the Activation Analysis Report. These results do not include the radioactivity from surface contamination. The results presented below for the package contents in their entirety and for the component identified as the worst case component include estimated surface contaminants.

5.1 Surface Contaminant Estimate

The basis for estimating surface contamination on the internals surfaces of the RPV and internals components is independent lab sample Z10449 dated 5/15/98 analyzed by Duke Engineering. This sample was used for the Steam Generators and the quantities and distribution of activity on the sample are considered representative of what would be present on RPV interior and reactor internals component surfaces.

The contaminants were estimated using these sample data in uCi/cm^2 and. the conservative estimate of 12,000 square feet of surface area. With a total surface contamination level of 17 uCi/cm^2, the resultant activity was about 190 curies. This includes 5.2 curies of Transuranics.

5.2 Package Content Waste Classification

Table 5-1 presents the 10 CFR Part 61 classification summary for the RPV and its contents. As shown the entire package is about 9 percent of the Class B limits and on average is Class B waste. The basis for this calculation is the displaced volume of the activated metal - 1,600 cubic feet. This volume excludes the RPV head, which was classified separately.

5.3 Worst Case Component Classification

The component within the package with the highest concentration of activation products is the Bottom Core Barrel Section. This component is a 41 inch segment of the Lower Core Barrel (see Attachment 3 Section 2.2.2 and, Addendum 1). This component is estimated to contain about 13,700 curies and weigh about 9,480 lbs.

Table 5-1

HNP Reactor Vessel and Internals
NRC Part 61 Classification Summary

Radionuclide	Total Activity Curies	Concentration uCi/cc	Concentration nCi/g	10 CFR 61 Table 1 Fraction	10 CFR 61 Table 2 Class A Frac.	10 CFR 61 Table 2 Class B Frac.	10 CFR 61 Table 2 Class C Frac
H-3	LLD	- - -	- - -				
C-14	3.75E+00	8.30E-02	- - -	1.04E-03			
Mn-54	2.78E+02	6.14E+00	- - -		8.77E-03		
Fe-55	1.87E+04	4.15E+02	- - -		5.92E-01		
Co-57	9.36E-02	2.07E-03	- - -		2.96E-06		
Co-60	1.88E+04	4.16E+02	- - -		5.94E-01		
Ni-59	2.12E+01	4.69E-01	- - -	2.13E-03			
Ni-63	2.84E+03	6.29E+01	- - -		1.80E+00	8.98E-02	8.98E-03
Sr-90	2.20E-02	4.86E-04	- - -		1.22E-02	3.24E-06	6.94E-08
Nb-94	5.90E-02	1.31E-03	- - -	6.53E-03			
Tc-99	1.91 E-02	4.22E-04	- - -	1.41E-04			
I-129	LLD	- - -	- - -				
Cs-137	LLD	- - -	- - -				
Np-237	6.02E-04	- - -	1.66E-03	1.66E-05			
Pu-238	2.54E-01	- - -	6.99E-01	6.99E-03			
Pu-239/240	8.12E-02	- - -	2.24E-01	2.24E-03			
Pu-241	4.54E+00	- - -	1.25E+01	3.57E-03			
Am-241	2.48E-01	- - -	6.84E-01	6.84E-03			
Cm-242	1.53E-03	- - -	4.21E-03	2.10E-07			
Cm-243/244	8.99E-02	- - -	2.48E-01	2.48E-03			
Totals	4.07E+04	9.00E+02	1.44E+01	0.03	3.00	0.09	0.01

Waste Weight 800,000lb
3.63E+08g
Waste Volume 1 .60E+03ft^3
4.52E+07cc

Definitions:
LLD = Lower Limit of Detection

Table 5-2 presents the 10 CFR Part 61 classification summary for the Bottom Core Barrel Sections. As shown these components are about 36 percent of the Class C limit. The basis for this calculation is the displaced volume of the activated metal - 18.9 cubic feet.

Since this is the highest specific activity component and concentration averaging will not be employed, the package contents will be shipped for disposal as Class C stable waste.

Table 5-2

HNP Bottom Core Barrel Section (41 inches)
NRC Part 61 Classification Summary

Radionuclide	Total Activity Curies	Concentration uCi/cc	Concentration nCi/g	10 CFR 61 Table 1 Class C Frac.	10 CFR 61 Table 2 Class C Frac
H-3	LLD	- - -	- - -		
C-14	1.36E+00	2.53E+00	- - -	3.16E-02	
Mn-54	7.81E+01	1.46E+02	- - -		
Fe-55	5.43E+03	1.01E+04	- - -		
Co-57	2.01E-03	3.76E-03	- - -		
Co-60	7.17E+03	1.34E+04	- - -		
Ni-59	7.73E+00	1.44E+01	- - -	6.56E-02	
Ni-63	1.04E+03	1.93E+03	- - -		2.76E-01
Sr-90	4.72E-04	8.81E-04	- - -		1.26E-07
Nb-94	2.31E-02	4.31E-02	- - -	2.15E-01	
Tc-99	5.13E-03	9.58E-03	- - -	3.19E-03	
I-129	LLD	- - -	- - -		
Cs-137	LLD	- - -	- - -		
Np-237	1.29E-05	- - -	3.01E-03	3.01E-05	
Pu-238	5.45E-03	- - -	1.27E+00	1.27E-02	
Pu-239/240	1.74E-03	- - -	4.06E-01	4.06E-03	
Pu-241	9.75E-02	- - -	2.27E+01	6.48E-03	
Am-241	5.33E-03	- - -	1.24E+00	1.24E-02	
Cm-242	3.28E-05	- - -	7.63E-03	3.82E-07	
Cm-243/244	1.93E-03	- - -	4.49E-01	4.49E-03	
Totals	1.37E+04	2.56E+04	2.60E+01	0.36	0.28

Waste Weight	9,480 lb
	4.30E+06 g
Waste Volume	1.89E+01 ft^3
	5.36E+05 cc

Definitions:

LLD = Lower Limit of Detection

6.0 REFERENCES

1. 10 CFR Part 61, Energy, Nuclear Regulatory Commission.

2. 49 CFR Part 107, Transportation, Hazardous Materials Program Procedures.

3. 49 CFR Part 173, Transportation, Shippers – General Requirements for Shipments and Packagings.

4. 49 CFR Part 393, Transportation, Parts and Accessories

5. AISC, American Institute of Steel Construction, “Steel Construction Manual,” Ninth Edition, 1989.

6. ANSI, N14.2, “Proposed American National Standard for Tiedowns for Truck Transport of Radioactive Materials,” January 7, 1999.

7. ANSI, N14.23, “Draft American National Standard Design Basis for Resistance to Shock and Vibration of Radioactive material Packages Greater than One Ton in Truck Transport,” May 1980.

8. ANSI, 14.24, “American National Standard for Highway Route Controlled Quantities of Radioactive Material-Domestic Barge Transport,” 1985.

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IO. ASME/ANSI, NQA-1, Subpart 2.15, “Quality Assurance Requirements for Hoisting, Rigging and Transporting of Items for Nuclear Power Plants,” 1997 Edition.

11. AWS, D1 . 1, “Structural Welding Code-Steel,” 1998.

12. Calculation No. 99-11-02, Tie-down Analysis for CY RVP.

13. Calculation No. 99-13-01, RVP Cradle Analysis.

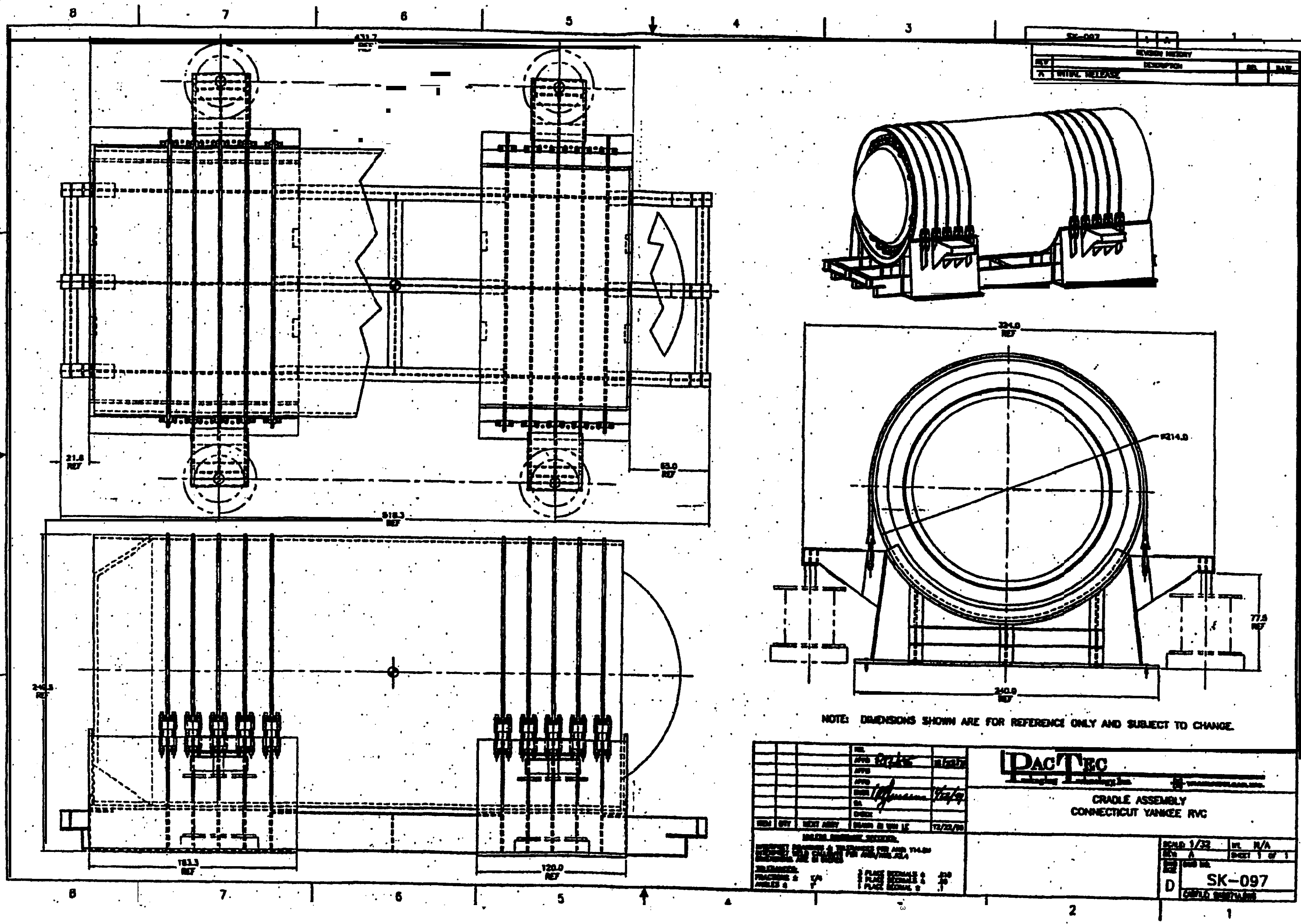
14. Calculation No. 99-11-06, Vibration Analysis for CY RVP.

15. Calculation No. 99-11-07, Impact Analysis for CY RVP.

16. Calculation No. 99-11-08, Stress Analysis for CY RVP.

APPENDIX A

HNP CRADLE ASSEMBLY DRAWING



SK-097
REVISION HISTORY
REV
DESCRIPTION
A
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21.8 REF
53.0 REF
518.3 REF
246.5 REF
153.3 REF
120.0 REF
324.0 REF
R214.0
77.5 REF
240.0 REF
NOTE: DIMENSIONS SHOWN ARE FOR REFERENCE ONLY AND SUBJECT TO CHANGE.
PACTEC
CRADLE ASSEMBLY
CONNECTICUT YANKEE RVC
SCALE 1/32
REV A
SHEET 1 OF 1
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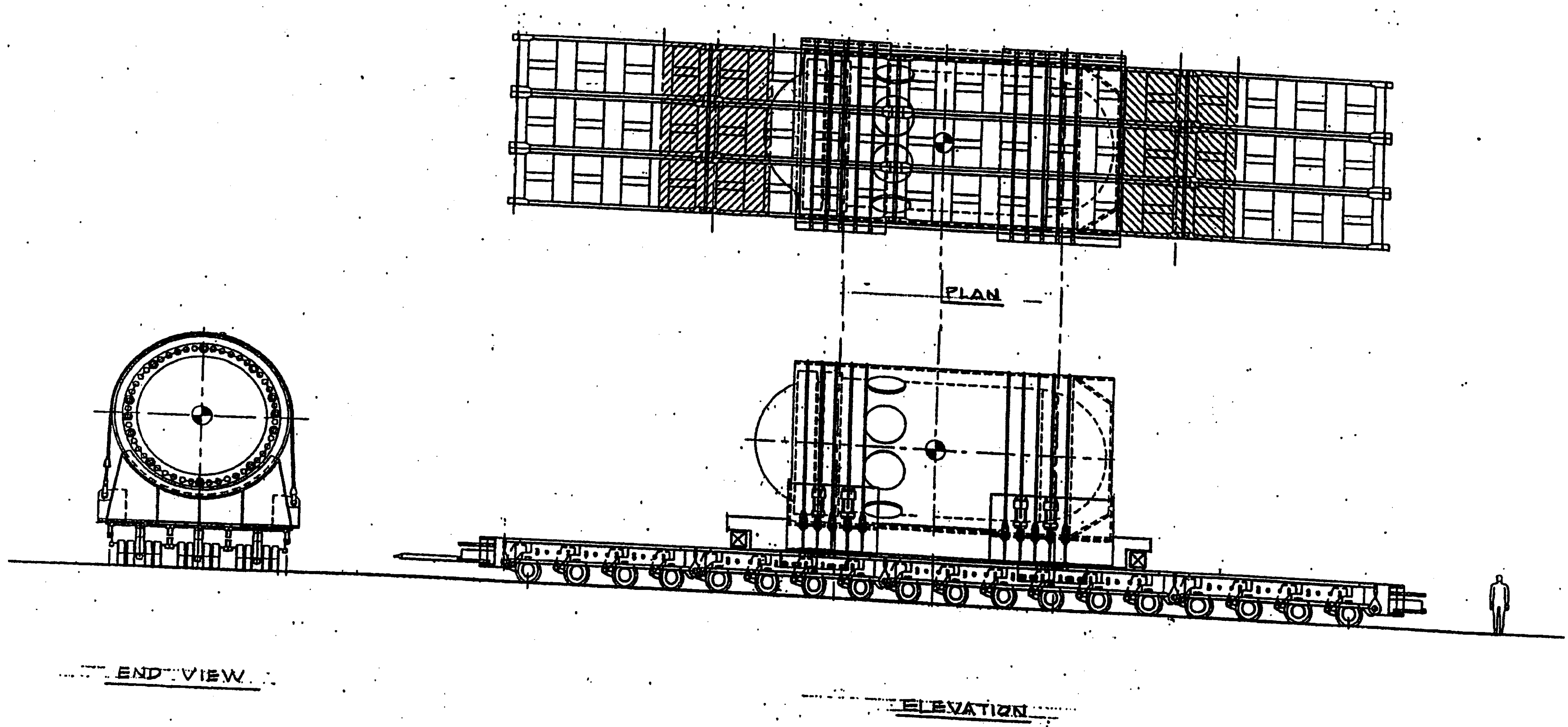
SK-097

APPENDIX B

SKETCH – TYPICAL LAND TRANSPORT CONFIGURATION



TYPICAL LAND TRANSPORT CONFIGURATION

[redacted]

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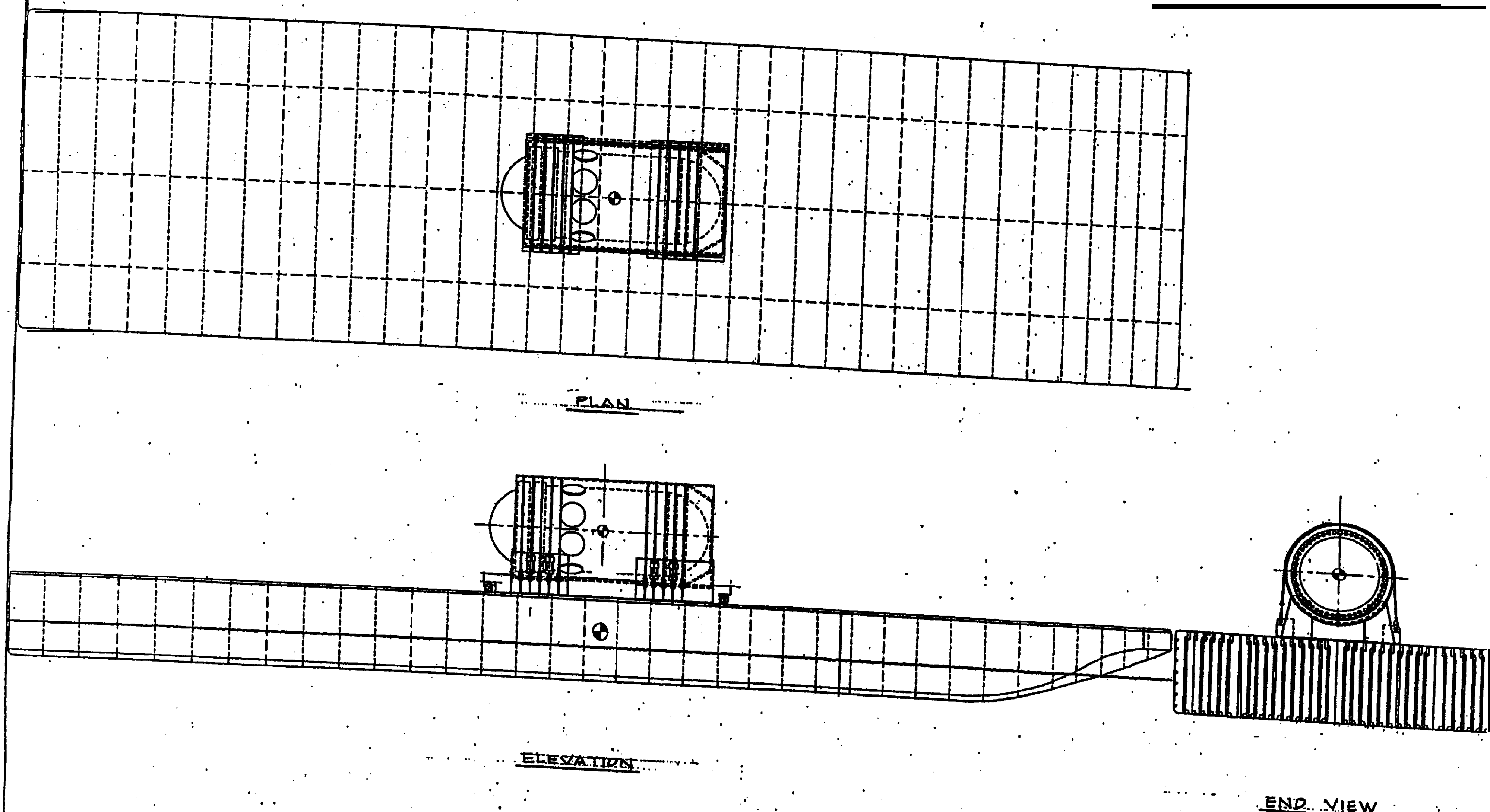
APPENDIX C

SKETCH – TYPICAL BARGE TRANSPORT CONFIGURATION



TYPICAL BARGE TRANSPORT CONFIGURATION

ATTACHMENT 3

HADDAM NECK

REACTOR VESSEL and INTERNALS

CHARACTERIZATION

Report WMG 9913-9007

and

ADDENDUM 1

HADDAM NECK

REACTOR VESSEL AND INTERNALS

CHARACTERIZATION

Report **WMG-9913-9007,** Rev. **1**

March 2000

WMG Project 9007E

Prepared for:

Bechtel Power Corporation

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HADDAM NECK
REACTOR VESSEL AND INTERNALS CHARACTERIZATION
REPORT WMG-9913-9007

WMG Inc.		16 Bank Street, Peekskill, NY 10566
Project Application	Copy No. N/A	Assigned To: N/A

APPROVALS			
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FOREWORD

This report summarizes the activation analysis work performed by WMG, Inc. to support the decommissioning of the Haddam Neck Plant for Bechtel Power Corporation. This work was performed under Task 2 of Subcontract 24265-TSC-200.

Revision 1 incorporated client comments and a correction to a minor computational error identified during an internal audit.

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1 .0 INTRODUCTION

In June 1999, Bechtel Power engaged WMG to support the Haddam Neck Decommissioning Project. One of several tasks assigned to WMG was to characterize and classify the Haddam Neck reactor vessel and internals. This report summarizes the results of WMG's analytical work to complete this Task. The results presented herein are based on analytical data as well as empirical data. The empirical data exists in the form of radiation level measurements. These results are preliminary because surface contaminant data, which is not yet available, has to be considered to quantify the transuranic and fission product content of the Low Level Radioactive Waste (LLRW) sent to the Barnwell disposal site. The scaling factors presented in this report, which estimate the hard to detect radionuclide concentrations important to classification under 10 CFR Part 61, are final and are used with the normalized activity results to classify each component.

Section 2 presents the analysis results and determines what components or pieces thereof can be disposed of within the reactor vessel package as LLRW in accordance with 10 CFR Part 61. Section 3 describes the method used to normalize the analytical results to empirical data and the benchmarking of the normalized analytical results.-

2.0 ESTIMATED COMPONENT RADIOACTIVITY AND CLASSIFICATION STATUS

The ANISN computer program was used in conjunction with ORIGEN2 activation analysis results to determine scaling factors as a function of component type and locale within the reactor vessel. The ORIGEN2 results are then normalized to the thermal shield radiation level measurements as discussed in section 3. These normalized ORIGEN2 results are used to estimate component activity and 10 CFR Part 61 classification status. This section presents the normalized ORIGEN2 estimated activity for each component of interest.

2.1 Overview

Table 2-1 summarizes the results in terms of total activity, Co-60 activity, and 10 CFR Part 61 classification status. The estimated activity for all components, which includes the reactor vessel, internals, and vessel insulation, is approximately 809,000 curies as of September I, 2000. This date reflects the earliest anticipated shipping date of the reactor vessel package. The core baffle assembly contains approximately 632,000 curies, or 78% of the total activity. The components are separated into two categories in Table 2-1 below. These categories are GTCC components and LLRW components. The GTCC components will have to be segmented and stored on site, while the LLRW components will be disposed of intact within the reactor vessel package. The GTCC components are the core baffle assembly, the lower core plate, and an 89 inch section of the lower core barrel that resides in the active fuel region. The remainder of the components meets all 10 CFR Part 61 requirements for disposal as LLRW.

The GTCC waste consists of about 37,400 lbs. of activated metal and contains approximately 769,000 curies. The LLRW internals that will be disposed of within the reactor vessel package consist of approximately 35,800 curies. When these internals are combined with the reactor vessel itself and the insulation is placed within the reactor vessel, the reactor vessel package will consist of 939,000 lbs. of activated metal containing approximately 40,500 curies as of the anticipated disposal date. The eight nozzles will be removed to reduce the reactor vessel outer diameter. Therefore, the weight of the reactor vessel may be reduced by the weight of the nozzles not placed within the reactor vessel, about 7,600 lbs. apiece. The specific characteristics of each component in Table 2-1 are discussed in detail below.

TABLE 2-1
Haddam Neck Component Activity and Part 61 Table

Component Name	Total Weight (lbs.)	Activity (Ci)	Co-60 Activity (Ci)	10 CFR Part 61 Classification Status	Drawing Numbers
Greater Than Class C Waste					
Core Baffle Assembly*	1.43E+04	6.32E+05	2.77E+05	GTCC	NU 16103-29300, Sh. 3A & 7
Center Core Barrel Section (Core Region, 89")	1.87E+04	9.96E+04	5.20E+04	GTCC	NU 16103-29300, Sh. 5A
Lower Core Plate	4.35E+03	3.67E+04	1.78E+04	GTCC	NU 16103-29300, Sh. 4A & 37
GTCC Subtotals	**3.74E+04**	**7.69E+05**	**3.47E+05**		
LLRW to be Disposed of intact Within the Reactor Vessel					
Above Core Regions					
Top Core Barrel Section (151")	3.64E+04	1.45E+04	7.57E+03	B	NU 16103-29300, Sh. 2A, 5A
Upper Core Plate	3.33E+03	1.12E+03	6.54E+02	B	NU 16103-29464, Sh. 6 & 10
Upper Internals (Excluding Upper Core Plate)	5.34E+04	3.16E+02	1.70E+02	A	NU 16103-29464, Sh. 12, 13, 15, 39 NU 16103-29300 Sh. 6A, 18
Uppers Subtotal	**9.31 E+04**	**1.59E+04**	**8.39E+03**		
Below Core Regions					
Lower Internals (excluding GTCC Lower Core Plate)	3.52E+04	6.14E+03	2.50E+03	B	NU 16103-29300, Sh. 9A, 17, 21, 27, 46, 47, 48, 49, 67, 76; NU 16103-29464 Sh. 7
Bottom Core Barrel Section (41")	9.48E+03	1.37E+04	7.17E+03	C	NU 16103-29300, Sh. 5A
Lowers Subtotal	**4.47E+04**	**1.99E+04**	**9.67E+03**		
LLRW internals Subtotal	**1.38E+05**	**3.58E+04**	**1.81 E+04**		
Reactor Pressure Vessel					
Vessel Cladding**	1.02E+04	4.70E+02	2.25E+02	B	NU 16103-29305 Sh. 1, CE 231-377-10, CE 231-371-9
Reactor Vessel***	7.87E+05	4.24E+03	3.75E+02	A	NU 16103-29305 Sh. 1, CE 231-377-10, CE 231-371-9
Vessel Insulation	3.82E+03	1.47E+01	8.22E+00	A	NU 16103-29198, Sh. 5
Subtotal	**8.01 E+05**	**4.72E+03**	**6.08E+02**		
Total for Vessel Package	**9.39E+05**	**4.05E+04**	**1.87E+04**		

*Includes Formers

**Includes Reactor Vessel & Closure Head Cladding

*** Includes Closure Head

2.2 Component Radioactivity

Individual component estimated activities are decay corrected to September 1, 2000, which is the earliest anticipated date to commence shipping operations. Components located more than a foot above and below the active core region are included in the upper internals and the lower internals. These components have relatively low specific activities and will be disposed of within the reactor vessel package. It is assumed that the surface contaminants will have a negligible impact on the characterization results, although the surface. 'contaminants could result in contact dose rates in the 1 to 3 R/hr range. This projection is based on the empirical data from Yankee Rowe.

The only activities presented in this report for the components of interest are radionuclides with half-lives greater than 90 days. This is done because of the relatively long cooling time between the end of irradiation and the anticipated disposal date. As a result, the short-lived activation products (i.e., Co-58, Cr-51, and Fe-59) were considered to be negligible.

2.2.1 Upper Internals

The upper internals consist of the upper support plate and deep beam support, the upper support column assembly, the guide tubes, the upper core plate, and the reactor closure head penetrations. This region extends upward from the top of the fuel assemblies. The upper core plate holds the fuel assemblies in place. The upper support columns and guide tubes are attached to the upper core plate and extend upward to the upper support plate assembly, consisting of deep beam support and upper support plate. The reactor closure head penetrations consist of the shroud tubes, the core deluge ports, and the thermocouple column assemblies that extend above the upper support plate to the closure head. The upper internals, including the upper core plate, have a total weight of about 56,750 lbs. and contain approximately 1,430 curies. Because of its proximity to the active fuel, the upper core plate is broken out separately from the rest of the upper internals.

The upper core plate has a weight of 3,330 lbs. and contains approximately 1,120 curies. The radionuclide content and distribution, as well as the 10 CFR Part 61 Status, for the upper core plate is presented in Table 2-2. The upper core plate is Class B waste and will be disposed of intact within the reactor vessel package.

The remaining upper internals have a weight of about 53,400 lbs. and contain approximately 320 curies. These components are Class A waste and will be disposed of within the reactor vessel package. The radionuclide content and distribution, as well as the 10 CFR Part 61 status, for the remaining upper internals is presented below in Table 2-3.

TABLE 2-2
Upper Core Plate Characterization Results

Component: Upper Core Plate	
Component Weight (lb.):	3.33E+03
Total Activity Curies	1.12E+03
Co-60 Activity Curies	6.54E+02
Part 61 Table 1 B Fraction	0.76
Part 61 Table 2 B Fraction	0.49

Nuclide	Curies/g	Estimated Curies	Scaling Factors
c 14	5.83E-08	8.81E-02	1.35E-04
Mn 54	6.13E-06	9.25E+00	1.41E-02
Fe 55	2.58E-04	3.89E+02	5.95E-01
Co 60	4.33E-04	6.54E+02	1.00E+00
Ni 59	3.18E-07	4.80E-01	7.34E-04
Ni 63	4.31E-05	6.51E+01	9.96E-02
Nb 94	1.44E-09	2.17E-03,.	3.32E-06
Tc 99	3.67E-10	5.55E-04	8.48E-07
Totals	**7.40E-04**	**1.12E+03**	

TABLE 2-3
Upper Internals Characterization Results

Component:	Upper Internals (excluding upper core plate)		
Component Weight (lb.):		5.34E+04	
Total Activity Curies		3.16E+02	
Co-60 Activity Curies		1.70E+02	
Part 61 Table 1 A Fraction		0.01	
Part 61 Table 2 A Fraction		0.31	
Nuclide	**Curies/g**	**Estimated Curies**	**Scaling Factor!**
C 14	. 9.94E-10	2.41E-02	1.42E-04
Mn 54	8.60E-08	2.08E+00	1.23E-02
Fe 55	5.16E-06	1.25E+02	7.36E-01
Co 60	7.01E-06	1.70E+02	1.00E+00
Ni 59	5.61E-09	1.36E-01	8.00E-04
Ni 63	7.53E-07	1.83E+01	1.07E-01
Nb 94	2.14E-11	5.18E-04	3.05E-06
Tc 99	5.26E-12	1.27E-04	7.50E-07
Totals	**1.30E-05**	**3.16E+02**	

2.2.2 Core Barrel Assembly

The core barrel assembly is 281 inches long and consists of the upper core barrel (107 inches) and the lower core barrel (174 inches). The assembly weighs about 64,600 lbs. and contains approximately 127,800 curies. The upper core barrel is well above the core region, while a major portion of the lower core barrel is adjacent to the core region. Thus, a portion of the lower core barrel is GTCC waste and has to be segmented and stored on site. Based on the results of this analysis, the core barrel assembly can be divided into three sections: a 151 inch long top section comprised of the entire upper core barrel plus 44 inches of the lower core barrel, an 89 inch long center section comprised of the mid-portion of the lower core barrel, and a 41 inch bottom section comprised of the lower portion of the lower core barrel. The top and bottom sections will be disposed of within the reactor vessel package. The center core barrel section will be segmented and stored on site as GTCC waste.

The top core barrel section is 151 inches long, weighs about 36,400 lbs., and contains approximately 14,500 curies. The component activities, 10 CFR Part 61 status, and scaling factors are summarized in Table 2-4 below. This component is Class B waste and will be disposed of within the reactor vessel package.

TABLE 2-4
Top Core Barrel Section Characterization Results

Component:	**Top Core Barrel Section (upper 151 inches)**		
Component Weight (lb.):		3.64E+04	
Total Activity Curies		1.45E+04	
Co-60 Activity Curies		7.57E+03	
Part 61 Table 1 B Fraction		0.87	
Part 61 Table 2 B Fraction		0.76	
Nuclide	**Curies/g**	**Estimated Curies**	**Scaling Factors**
c 14	8.68E-08	1.43E+00	1.89E-04
Mn 54	4.99E-06	8.24E+01	1.09E-02
Fe 55	3.48E-04	5.73E+03	7.58E-01
Co 60	4.59E-04	7.57E+03	1.00E+00
Ni 59	4.94E-07	8.16E+00	1.08E-03
Ni 63	6.62E-05	1.09E+03	1.44E-01
Nb 94	1.48E-09	2.43E-02	3.22E-06
Tc 99	3.28E-10	5.42E-03	7.16E-07
Totals	**8.78E-04**	**1.45E+04**	

The center section (next 89 inches) of the core barrel weighs about 18,700 lbs. and contains approximately 99,600 curies. The component activities, 10 CFR Part 61 status, and scaling factors are summarized in Table 2-5. Because this section is adjacent to the active core region, it is GTCC waste and will be segmented and stored on site.

TABLE 2-5
Center Core Barrel Section Characterization Results

Component:	Center Core Barrel Section (89 inches)
Component Weight (lb.):	1.87E+04
Total Activity Curies	9.96E+04
Co-60 Activity Curies	5.20E+04
Part 61 Table 1 C Fraction	1.16
Part 61 Table 2 C Fraction	1.01

Nuclide	Curies/g	Estimated Curies	Scaling Factors
C 14	1.16E-06	9.84E+00	1.89E-04
Mn 54	6.66E-05	5.66E+02	1.09E-02
Fe 55	4.64E-03	3.94E+04	7.58E-01
Co 60	6.12E-03	5.20E+04	1.00E+00
Ni 59	6.60E-06	5.61E+01	1.08E-03
Ni 63	8.83E-04	7.51E+03	1.44E-01
Nb 94	1.97E-08	1.67E-01	3.22E-06
Tc 99	4.38E-09	3.72E-02	7.16E-07
Totals	**1.17E-02**	**9.96E+04**	

The bottom section of the core barrel is the remaining 41 inches. It weighs about 9,500 lbs. and contains approximately 13,700 curies. The component activities, 10 CFR Part 61 status, and scaling factors are summarized in Table 2-6. This component is Class C waste and will be disposed of within the reactor vessel package.

TABLE 2-6
Bottom Core Barrel Section Characterization Results

Component: Bottom Core Barrel Section (41 inches)			
Component Weight (lb.):		9.48E+03	
Total Activity Curies		1.37E+04	
Co-60 Activity Curies		7.17E+03	
Part 61 Table 1 C Fraction		0.31	
Part 61 Table 2 C Fraction		0.28	
Nuclide	**Curies/g**	Estimated Curies	**Scaling** Factors
c 14	3.15E-07	1.36E+00	1.89E-04
Mn 54	1.81E-05	7.81E+01	1.09E-02
Fe 55	1.26E-03	5.43E+03	7.58E-01
Co 60	1.67E-03	7.17E+03	1.00E+00
Ni 59	1.80E-06	7.73E+00	1.08E-03
Ni 63	2.41E-04	1.03E+03	1.44E-01
Nb 94	5.36E-09	2.31E-02	3.22E-06
Tc 99	1.19E-09	5.13E-03	7.16E-07
Totals	**3.19E-03**	**1.37E+04**	

2.2.3 Core Baffle Assembly

The core baffle assembly is divided into two regions in the radial transport model. These regions are the baffle plate region and the baffle former region. This is done because of the large difference in the water to metal fractions in the two regions. The core baffle is constructed of 0.5 inch thick baffle plates and is held in place by the baffle formers with an offset of about 0.675 inches from the outer fuel assemblies. The activation analysis results for these two regions have been combined in this report since it will be considered to be a single component for storage as GTCC waste. The component activities, 10 CFR Part 61 status, and scaling factors for the entire assembly are summarized in Table 2-7 below. The core baffle assembly has a total activity of approximately 632,000 curies and weighs 14,300 lbs. The core baffle assembly is well above Class C limits and will be segmented and stored on site as GTCC waste.

TABLE 2-7
Core Baffle Assembly Characterization Results

Component: Core Baffle Assembly			
Component Weight (lb.):		1.43E+04	
Total Activity Curies		6.32E+05	
Co-60 Activity Curies		2.77E+05	
Part 61 Table 1 C Fraction		8.98	
Part 61 Table 2 C Fraction		8.47	
Nuclide	**Curies/g**	**Estimated Curies**	**Scaling Factors**
c 14	1.11E-05	7.24E+01	2.61E-04
Mn 54	6.30E-04	4.09E+03	1.48E-02
Fe 55	4.66E-02	3.03E+05	1.09E+00
Co 60	4.27E-02	2.77E+05	1.00E+00
Ni 59	4.17E-05	2.71E+02	9.78E-04
Ni 63	7.41E-03	4.81E+04	1.73E-01
Nb 94	1.57E-07	1.02E+00	3.67E-06
Tc 99	2.81E-08	1.83E-01	6.59E-07
Totals	**9.74E-02**	**6.32E+05**	

2.2.4 Lower Internals

The lower internals includes all the components that support the fuel and upper internals. The lower internals include the lower core plate, core support columns, core support, access port plug assembly, intermediate plate, instrumentation guide tubes, tie plate supports, instrumentation guide tube tie plate, secondary core support baseplate, and secondary core support assembly. This region extends from the bottom of the fuel assemblies to the reactor vessel bottom head. The lower internals, including the lower core plate, has a total weight of about 39,600 lbs. and contains approximately 42,800 curies. The lower core plate contains approximately 36,700 curies, or 86% of the total activity. This is a result of the lower core plate being located close the active fuel region and requires the lower core plate to be examined separately from the rest of the lower internals assembly.

2.2.4.1 Lower Core Plate

The lower core plate is a 1.5 inch thick, 130.5 inch diameter plate that supports the fuel assemblies and weighs 4,350 lbs. including the fuel assembly guide pins. The results of the characterization and 10 CFR Part 61 status of the lower core plate are presented below in Table 2-8. The lower core plate is above Class C limits and will be segmented and stored on site as GTCC waste.

TABLE 2-8
Lower Core Plate Characterization Results

Component: Lower Core Plate			
Component Weight (lb.):		4.35E+03	
Total Activity Curies		3.67E+04	
Co-60 Activity Curies		1.78E+04	
Part 61 Table 1 C Fraction		1.70	
Part 61 Table 2 C Fraction		1.74	
Nuclide	**Curies/q**	**Estimated Curies**	**Scaling Factors**
c 14	1.98E-06	3.91E+00	2.20E-04
Mn 54	8.45E-05	1.67E+02	9.37E-03
Fe 55	7.93E-03	1.57E+04	8.79E-01
Co 60	9.02E-03	1.78E+04	1.00E+00
Ni 59	1.14E-05	2.24E+01	1.26E-03
Ni 63	1.52E-03	3.00E+03	1.69E-01
Nb 94	2.68E-08	5.28E-02	2.97E-06
Tc 99	5.26E-09	1.04E-02	5.82E-07
Totals	**1.86E-02**	**3.67E+04**	

2.2.4.2 Remaining Lower Internals

The remaining lower internals have a weight of 35,200 lbs. and contain approximately 6,140 curies. The component activities, 10 CFR Part 61 status, and scaling factors for the entire assembly are summarized in Table 2-9 below. The remaining lower internals are Class B waste and will be disposed of within the reactor vessel package.

TABLE 2-9
Remaining Lower Internals Characterization Results

Component: Remaining Lower Internals			
Component Weight (lb.):		3.52E+04	
Total Activity Curies		6.14E+03	
Co-60 Activity Curies		2.50E+03	
Part 61 Table 1 B Fraction		0.28	
Part 61 Table 2 B Fraction		0.40	
Nuclide	**Curies/g**	**Estimated Curies**	**Scaling Factors**
C 14	4.42E-08	7.06E-01	2.83E-04
Mn 54	5.87E-07	9.37E+00	3.75E-03
Fe 55	1.92E-04	3.07E+03	1.23E+00
Co 60	1.56E-04	2.50E+03	1.00E+00
Ni 59	2.65E-07	4.23E+00	1.69E-03
Ni 63	3.49E-05	5.58E+02	2.23E-01
Nb 94	3.51E-10	5.60E-03	2.24E-06
Tc 99	3.97E-11	6.34E-04	2.54E-07
Totals	**3.84E-04**	**6.14E+03**	

2.2.5 Reactor Vessel Assembly

The reactor vessel assembly consists of the reactor vessel, the reactor closure head, the stainless steel vessel cladding, and the vessel mirror insulation. The cladding region was separated from the vessel wall region because they are fabricated from two different materials, which are stainless steel and carbon steel, respectively. This causes the activation of these two regions to differ significantly. The curie content for the reactor vessel, cladding and insulation regions adjacent to the active fuel is determined using ORIGEN2. Based on previous analytical models for Trojan and Maine Yankee, and empirical data from Haddam Neck, it is assumed 90 percent of the total activity of the vessel components is located adjacent to the active fuel region. The remaining activity is located

+ 2 feet of the active fuel. The results for the reactor vessel assembly are summarized below.

2.2.5.1 Reactor Vessel and Closure Head

The reactor vessel and closure head have a combined height of 463 ½ inches. The vessel itself has an outer diameter of 175 9/16 inches and a wall thickness of 10 25/32 inches. The closure head is mounted on the vessel flange that has an outer diameter of 189 inches. These components have a combined weight of 787,000 lbs. and a total activity of approximately 4,240 curies. The closure head has negligible activation activity. The results of the characterization and 10 CFR Part 61 status of these components are presented below in Table 2-10. The reactor vessel and closure head are Class A waste.

TABLE 2-10
Reactor Vessel Characterization Results

Component:	Reactor Vessel and Closure Head
Component Weight (lb.):	7.87E+05
Total Activity Curies	4.24E+03
Co-60 Activity Curies	3.75E+02
Part 61 Table 1 A Fraction	< 0.01
Part 61 Table 2 A Fraction	0.15

Nuclide	Curies/g	Estimated Curies	Scaling Factors
c 14	2.63E-10	9.38E-02	2.50E-04
Mn 54	2.62E-07	9.34E+01	2.49E-01
Fe 55	1.05E-05	3.75E+03	9.99E+00
Co 60	1.05E-06	3.75E+02	1.00E+00
Ni 59	4.71E-10	1.68E-01	4.49E-04
Ni 63	6.42E-08	2.29E+01	6.12E-02
Nb 94	7.52E-12	2.69E-03	7.16E-06
Tc 99	1.99E-11	7.10E-03	1.89E-05
Totals	**1.19E-05**	**4.24E+03**	

2.2.5.2 Stainless Steel Vessel Cladding

The vessel cladding is a stainless steel weld deposited along the entire inside surface of the reactor vessel and closure head. It has a nominal thickness of 5/32 inch. The total weight of the cladding is about 10,200 lbs. with a total activity of approximately 470 curies. Estimated activities, 10 CFR Part 61 status and scaling factors are summarized below in Table 2-1 1 for the vessel cladding. The cladding material, an integral part of the reactor vessel, is Class B waste and is acceptable for disposal.

TABLE 2-1 1
Vessel Cladding Characterization Results

Component: Vessel Cladding			
Component Weight (lb.):		1.02E+04	
Total Activity Curies		4.70E+02	
Co-60 Activity Curies		2.25E+02	
Part 61 Table 1 B Fraction		0.09	
Part 61 Table 2 B Fraction		0.10	
Nuclide	**Curies/g**	**Estimated Curies**	**Scaling Factors**
C 14	1.09E-08	5.04E-02	2.24E-04
Mn 54	4.07E-07	1.88E+00	8.37E-03
Fe 55	4.39E-05	2.03E+02	9.02E-01
Co 60	4.86E-05	2.25E+02	1 .00E+00
Ni 59	6.74E-08	3.12E-01	1.39E-03
Ni 63	8.59E-06	3.97E+01	1.77E-01
Nb 94	1.31E-10	6.07E-04	2.70E-06
Tc 99	2.42E-11	1.12E-04	4.98E-07
Totals	1.02E-04	4.70E+02	

2.2.5.3 Vessel Mirror Insulation

The vessel mirror insulation surrounds the reactor vessel and is constructed of three thin sheets of stainless steel. There is a ½ inch air gap between the insulation and the outside of the reactor vessel. The insulation is 3 inches thick and an estimated density of 3.12 lb/in*. The mirror insulation on the bottom head of the vessel was replaced with blanket insulation and therefore is not included. The total weight of the cylindrical section of the insulation is about 3,800 lbs. with a total activity of approximately 15 curies. Component estimated activities, 10 CFR Part 61 status, and scaling factors are presented below in Table 2-12. The insulation is Class A waste and will be removed and disposed of within the reactor vessel package.

TABLE 2-1 2
Mirror Insulation Characterization Results

Component: Mirror Insulation	
Component Weight (lb.):	3.82E+03
Total Activity Curies	1.47E+01
Co-60 Activity Curies	8.22E+00
Part 61 Table 1 A Fraction	0.01
Part 61 Table 2 A Fraction	0.17

Nuclide	Curies/g	Estimated Curies	Scaling Factors
c 14	4.10E-10	7.10E-04	8.64E-05
Mn 54	8.58E-08	1.49E-01	1.81E-02
Fe 55	3.33E-06	5.77E+00	7.02E-01
Co 60	4.75E-06	8.22E+00	1.00E+00
Ni 59	2.24E-09	3.88E-03	4.72E-04
Ni 63	3.13E-07	5.42E-01	6.59E-02
Nb 94	1.01 E-l 1	1.75E-05	2.13E-06
Tc 99	2.63E-12	4.55E-06	5.53E-07
Totals	**8.48E-06**	**1.47E+01**	

3.0 NORMALIZATION AND BENCHMARKING TO SURVEY RESULTS

The characterization results for the reactor vessel and internals were normalized to measured survey results obtained on the thermal shield after its removal in 1990. The normalized results were then benchmarked relative to surveys taken on the reactor vessel exterior on contact with the mirror insulation in September of 1998. Copies of the actual survey results are included in Appendix A.

The thermal shield was a 157 inch high, 4 inch thick hollow cylinder located around the reactor core between the lower core barrel and the reactor vessel wall. The thermal shield was removed after cycle 15 of operation. A 120 degree arced section of the thermal shield was surveyed on February 12, 1990. WMG used the 1 foot survey results in conjunction with a detailed QAD-CGGP-A model to normalize the ORIGEN2 activation results. The normalization to the thermal shield was based on the average survey results performed over a quarter of the thermal shield because of quarter core symmetry. This accounted for azimuthal peaking in the measured survey results.

The activation analysis for the reactor vessel with the internals was also benchmarked to external surveys taken on September IO, 1998. Detailed QAD-CGGP-A models were constructed which accounted for the distribution of the activity through the reactor vessel wall. Separate models were developed to determine the contribution from the reactor vessel wall source and mirror insulation. The calculated dose rate on contact with the mirror insulation at the core centerline as of September 10, 1998 was 7.99 R/hr. This result is within 6% of the measured survey of 8.5 R/hr, which validates the characterization results.

APPENDIX A

THERMAL SHIELD AND EXTERNAL REACTOR VESSEL SURVEYS

TITLE T/S OUTSIDE	RWP NO.: N/A	CONTROL # N/A	DATE 2/12/90	TIME 10:00
REASON FOR SURVEY.. . . . ROUTINE (D, W, M) ☐ PRE-JOB ☐ POST-JOB ☐		TECHNICIAN NEAL/THOMAS	REACTOR POWER-- N/A	

RADIATION		CONTAMINATION	
$\beta\gamma$	a	$\beta\gamma$	a
INST. TYPE PR-2	INST. TYPE N/A	INST. TYPE N/A	INST. TYPE N/A
SERIAL TYPE 141	SERIAL TYPE N/A	SERIAL TYPE N/A	SERIAL TYPE N/A
PROBE TYPE HIGH RANGE	PROBE TYPE N/A	PROBE TYPE N/A	PROBE TYPE N/A
CAL. DUE DATE 6/90	CAL. DUE DATE N/A	(CAL. DUE DATE N/A	(CAL. DUE DATE N/A

DOSE RATE READINGS ARE IN MREM/HR AT WAIST LEVEL UNLESS OTHERWISE SPECIFIED.
CIRCLED NUMBERS 4 INDICATE SMEAR LOCATIONS.
CIRCLED SMEAR LOCATIONS AND NUMBER 4 5000 INDICATE CONTAMINATION LEVELS IN DPM/100CM*2.
* READINGS INSIDE TRIANGLE /\ INDICATE NEUTRON RADIATION IN MREM/HR AT WAIST LEVEL.

STELLITE TOP - 600 R/hr 1' ABOVE - 40000 R/hr
MID - 45000 R/hr 1' BELOW - 30000 R/hr

111°	1'	3'	5	7	9	11'	12
2	50 R/hr X	60 R/hr X	150 R/hr X	250 R/hr X	400 R/hr X	30 R/hr X	30 R/hr X
5	120 R/hr X	130 R/hr X	300 R/hr X	900 R/hr X	800 R/hr X	100 R/hr X	100 R/hr X
a	100 R/hr X	130 R/hr X	350 R/hr X	900 R/hr X	700 R/hr X	140 R/hr X	130 R/hr X
12	80 R/hr X	100 R/hr X	200 R/hr X	400 R/hr X	500 R/hr X	80 R/hr X	100 R/hr X

☐ ALL SMEARS LESS THAN 1000 DPM/100 CM*2 $\beta\gamma$.	☐ 25% OF ALL SMEARS LESS THAN 20 DPM/100 CM*2.
HEALTH PHYSICS SUPERVISOR/DESIGNEE	

THIS IS A REPRODUCTION OF THE ORIGINAL FROM CALCULATION CYC-002, PAGE 109 (12 INCH DATA). DATA HAS BEEN TRANSFERRED FROM THE ORIGINAL DOCUMENT. NO NEW DATA WAS OBTAINED FOR THIS REPRODUCTION.

TITLE T/S INSIDE	RWP NO.: N/A	CONTROL # N/A	DATE 2/12/90	TIME 10:00
REASON FOR SURVEY..... ROUTINE (D, W, M) ☐ PRE-JOB ☐ POST-JOB ☐		TECHNICIAN NEAL/THOMAS	REACTOR POWER N/A	

RADIATION		CONTAMINATION	
βγ	a	βγ	a
INST. TYPE PR-2	INST. TYPE N/A	INST. TYPE N/A	INST. TYPE N/A
SERIAL TYPE 141	SERIAL TYPE N/A	SERIAL TYPE N/A	SERIAL TYPE N/A
PROBE TYPE HIGH RANGE	PROBE TYPE N/A	PROBE TYPE N/A	PROBE TYPE N/A
CAL. DUE DATE 6/90	CAL. DUE DATE N/A	CAL. DUE DATE N/A	CAL. DUE DATE N/A

DOSE RATE READINGS ARE IN MREM/HR AT WAIST LEVEL UNLESS OTHERWISE SPECIFIED.
CIRCLED NUMBERS (4) INDICATE SMEAR LOCATIONS.
CIRCLED SMEAR LOCATIONS AND NUMBER((4) 5000) INDICATE CONTAMINATION LEVELS IN DPM/100CM*2.
* READINGS INSIDE TRIANGLE /\ INDICATE NEUTRON RADIATION IN MREM/HR AT WAIST LEVEL.

111°	1'	3	5'	7	9	11'	12'
2	300 R/hr X	200 R/hr X	300 R/hr X	500 R/hr X	1500 R/hr X	200 R/hr X	150 R/hr X
5	700 R/hr X	800 R/hr X	1000 R/hr X	2400 R/hr X	3000 R/hr X	600 R/hr X	450 R/hr X
8	800 R/hr X	600 R/hr X	900 R/hr X	2500 R/hr X	4000 R/hr X	800 R/hr X	600 R/hr X
12	320 R/hr X	280 R/hr X	300 R/hr X	600 R/hr X	1000 R/hr X	250 R/hr X	200 R/hr X

☐ ALL SMEARS LESS THAN 1000 DPM/100 CM*2 βγ.	☐ 25% OF ALL SMEARS LESS THAN 20 DPM/100 CM*2.
HEALTH PHYSICS SUPERVISOR/DESIGNEE	

THIS IS A REPRODUCTION OF THE ORIGINAL FROM CALCULATION CYC-002, PAGE 1 10 (CONTACT DATA). DATA HAS BEEN TRANSFERRED FROM THE ORIGINAL DOCUMENT. NO NEW DATA WAS OBTAINED FOR THIS REPRODUCTION.

CONNECTICUT YANKEE ATOMIC POWER COMPANY

SURVEY RECORD FORM

COPY

Surveyor Name: (Printed)	Surveyor Name: (Signature)	Location:
Bill Hawkins / Brian Jones	B. Hawk / Brian Jones	Containment Building Reactor Vessel Cut Away View

Date: 9-10-98	Time: 1500	Log #: NA
Area #: 3324	Map #: RP882	Rev #: 00

INSTRUMENTS USED

MODEL/SERIAL#	CAL DUE	PROBE/SERIAL#	CAL DUE
E530 1026	12-98	← N/A	→
202A 3643	12-98	← N/A	→
APCE 70194	11-98	← N/A	→
←	→	N/A	→

REASON FOR SURVEY

- ☐ ROUTINE — N/A
- ☑ RWP — 46-2
- ☐ OTHER — N/A

Air Sample Control #: 09106 | Total DAC Fraction: .55

CONTAMINATION RESULTS (SMEARS) dpm/100cm²

β Ymda= 43.40 | α mda= 8.98

Smear #	Results β-γ	α	Smear #	Results β-γ	α
1	24272.9	99.0	11	7965.5	122.[illegible]
2	12906.2	138.6	Rx Vess. Mirror Insul.		
3	5454.5	89.1			
4	5408.7	42.9			
5	11493.8	79.2			
6	41095.9	531.6		N/A	
7	14079.5	92.4			
8	35285.0	544.8			
9	1229.3	<MDA			
10	1536.5	<MDA			

Smear Continuation Sheet Used: ☐ Yes ☑ No

NOTE 8

SURVEY PERFORMED IN 1 FOOT INCREMENTS BETWEEN THE Rx VESS. & N.S.T STARTING @ 1' BELOW THE TOP N.S.T

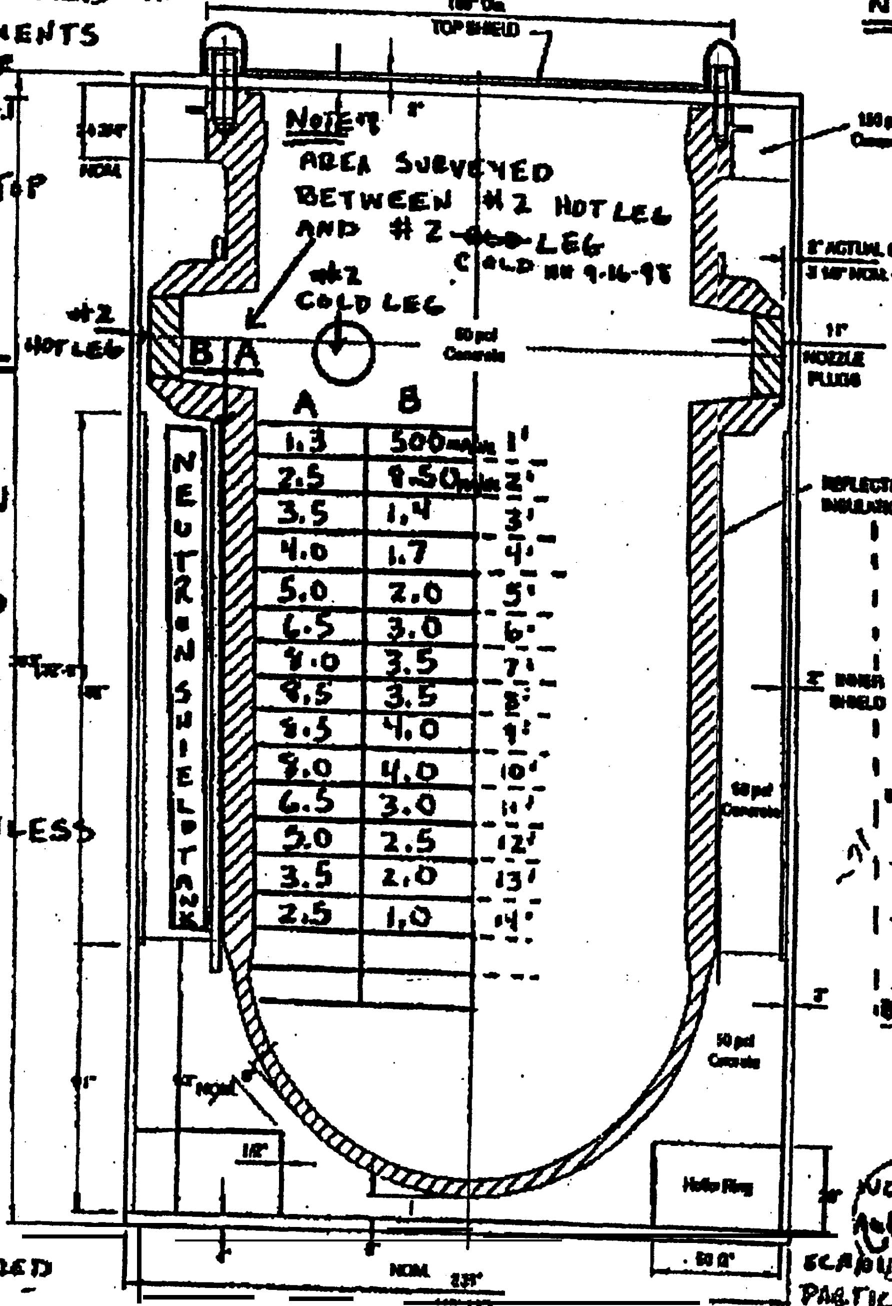
A	B
(#) CONTACT ON VESSEL MIRROR INSUL.	(#) CONTACT ON NEUTRON SHIELD TANK (N.S.T)

NOTE 8

ALL DOSE RATES ARE IN R/HR UNLESS OTHERWISE SPECIFIED

ENTERED 9/24/98

AREA POSTED PERMIT REQUIRED



A	B	
1.3	500 mR/hr	1'
2.5	9.50 mR/hr	2'
3.5	1.4	3'
4.0	1.7	4'
5.0	2.0	5'
6.5	3.0	6'
8.0	3.5	7'
8.5	3.5	8'
8.5	4.0	9'
8.0	4.0	10'
6.5	3.0	11'
5.0	2.5	12'
3.5	2.0	13'
2.5	1.0	14'

NOTE 8

OTHER DOSE RATES

\#100 mR/hr COLD LEG

\#120 mR/hr HOT LEG

100-250 mR/hr G/A

* 300 ON VESSEL mR/hr

SMEAR LOCATION

TOP OF NEUTRON SHIELD TANK

COLD LEG MIRROR INSUL.

NEUTRON SHIELD TANK COLD LEG

PAD INSUL. Rx VESS.

MIRROR INSUL. NEUT. SHIELD TANK

BTW Rx VESS & TANK

WORK AREA AND COWORKERS SCANNED FOR HOT PARTICLES.

Legend:

Dose rates are gamma and are in mR/hr, unless otherwise noted.

- Large Area Smear Location — HRA • High Radiation Area
- Air Sample Location — LHRA • Locked High Radiation Area
- Smear Location — HPA • Hot Particle Area
- Contact Exposure Rate — CA • Contaminated Area
- General Area Exposure Rate — HCA • High Contamination Area
- Indicates Boundary — ARA • Airborne Radioactivity Area
- RA • Radiation Area — RM • Radioactive Material
- Step Off Pad — RCA • Radiologically Controlled Area

HP MANAGER OR DESIGNEE REVIEW: R. L. DATE: 9-22-98

REPORT WMG-9913-9007

ADDENDUM 1

1.0 INTRODUCTION

This addendum to WMG–9913-9007 Rev. 1 dated March 2000, provides estimated surface contamination for the reactor vessel and internals based on steam generator data. Detailed characterization results are also provided for the worst case component, the bottom core barrel section (41 inches), along with the estimated results for the reactor vessel head.

2.0 ESTIMATED SURFACE CONTAMINATION

The surface contaminants on the vessel internals and internal 'surfaces of the reactor vessel body were estimated based on steam generator data. This data is used for estimating purposes only, and actual surface contamination samples will be obtained from the internals surfaces once the vessel head is removed. Based on detailed drawing take-offs performed for the Trojan reactor vessel package, the surface area for the vessel and internals was conservatively estimated at 12,000 ft^2. This surface area was used in conjunction with quantitative surface contamination data from Duke Engineering laboratory sample Z10449 dated 5/15/98. The total surface contamination was estimated at about 190 curies as shown in Table 1 below. The surface contaminants for the vessel head are addressed separately below.

3.0 BOTTOM CORE BARREL SECTION (41 INCHES)

The highest radionuclide concentrations that will be present in the reactor vessel package will be in the lower 41 inches of the bottom core barrel section. Characterization results for this component were broken out separately in Section 2.2.2 of WMG-9913-9007 Rev. 1 and are also provided in Table 2 below.

4.0 REACTOR VESSEL HEAD

The Reactor Pressure Vessel (RPV) head will be attached to the Canister and is thus included in the packaging. Prior to its attachment, the interior and exterior surfaces of the RPV head will be hydrolased to remove most of the removable surface contaminants.

To estimate the quantity of Class 7 (radioactive) material present on the head from surface contamination, the following conservative assumptions were made:

1. Prior to hydrolasing, the level of contamination on the interior surfaces is the same as that currently assumed for the reactor internals - about 17 uCi/cm^2. This level of contamination will also be present within the Control Rod Drive (CRD) penetrations.

2. Prior to hydrolasing, the level of contamination on the exterior surfaces will conservatively be IO percent of that on the interior surfaces - about 1.7 uCi/cm^2. This level of contamination will also be present within the stud bolt holes.

3. Hydrolasing both these areas prior to shipment will reduce the level of removable contamination a minimum factor of 10.

These assumptions result in an average estimated activity from surface contamination at the time of shipment of about 745 mCi. Of this amount, about 640 mCi (I .7 uCi/cm^2) is estimated for the interior surfaces and about 105 mCi (0.17 uCi/cm^2) is estimated for the head exterior surfaces. These levels are below SCO-II limits for removable beta/gamma surface contamination.

After hydrolasing and before coating with paint, swipe samples will be obtained from RPV head interior and exterior surfaces to perform final characterization. We do not expect the total activity from surface contaminants to be greater than the above estimate of 745 mCi.

Table 1

RPV and Internals Total Surface Contamination

Nuclide	Sample Data* (uCi/cm^2)	Total Activity (Ci)	Average Concentration (uCi/g)	A2 Value Curies	Type A Fraction
Np-237	5.40E-05	6.02E-04	1.66E-06	5.41E-03	1.11E-01
Pu-238	2.28E-02	2.54E-01	6.99E-04	5.41 E-03	4.69E+01
Pu-239,40	7.28E-03	8.12E-02	2.24E-04	5.41 E-03	1.50E+01
Pu-241	4.07E-01	4.54E+00	1.25E-02	2.70E-01	1 .68E+01
Am-241	2.23E-02	2.48E-01	6.84E-04	5.41 E-03	4.59E+01
Cm-242	1.37E-04	1.53E-03	4.21E-06	2.70E-01	5.66E-03
Cm-243,44	8.06E-03	8.99E-02	2.48E-04	8.11E-03	1.11E+01
TRU Subtotal	**4.68E-01**	**5.21 E+00**	**1.44E-02**		**1.36E+02**
Mn-54	8.17E-02	9.1 1E-01	2.51E-03	2.70E+01	3.37E-02
Co-57	8.40E-03	9.36E-02	2.58E-04	2.16E+02	4.34E-04
Co-60	1.23E+01	1 .37E+02	3.78E-01	1.08E+01	1.27E+01
Fe-55	3.49E+00	3.89E+01	1.07E-01	1.08E+03	3.60E-02
Ni-63	7.72E-01	8.61 E+00	2.37E-02	8.11 E+02	1.06E-02
Sr-90	1.97E-03	2.20E-02	6.05E-05	2.70E+00	8.13E-03
Subtotals	**1 .67E+01**	**1 .86E+02**	**5.12E-01**		**1.28E+01**
Totals	**17.1**	**191.0**	**5.26E-01**		**148.5**

RPV and Internals Estimates

Total Surface Area 12,000 ft^2

Total Weight* 8.00E+05 lb

*** Weight of RPV and Internals Without the Closure Head**

Note: Sample data taken from Duke sample ID #Z10449 dated 5/15/1998

Table 1 (Continued)

Bottom Core Barrel Section (41 inches)

Nuclide	Sample Data* (uCi/cm^2)	Total Activity (Ci)	Average Concentration (uCi/g)	A2 Value Curies	Type A Fraction
Np-237	5.40E-05	1.29E-05	3.01E-06	5.41 E-03	2.39E-03
Pu-238	2.28E-02	5.45E-03	1.27E-03	5.41E-03	1.01E+00
Pu-239,40	7.28E-03	1.74E-03	4.06E-04	5.41 E-03	3.23E-01
Pu-241	4.07E-01	9.75E-02	2.27E-02	2.70E-01	3.61 E-01
Am-241	2.23E-02	5.33E-03	1.24E-03	5.41 E-03	9.86E-01
Cm-242	1.37E-04	3.28E-05	7.63E-06	2.70E-01	1.22E-04
Cm-243,44	8.06E-03	1.93E-03	4.49E-04	8.11 E-03	2.38E-01
TRU Subtotal	**4.68E-01**	**1.12E-01**	**2.60E-02**		**2.92E+00**
Mn-54	8.17E-02	1.96E-02	4.55E-03	2.70E+01	7.25E-04
Co-57	8.40E-03	2.01 E-03	4.68E-04	2.16E+02	9.32E-06
Co-60	1.23E+01	2.95E+00	6.86E-01	1.08E+01	2.73E-01
Fe-55	3.49E+00	8.36E-01	1.94E-01	1 .08E+03	7.74E-04
Ni-63	7.72E-01	1.85E-01	4.30E-02	8.11 E+02	2.28E-04
Sr-90	1.97E-03	4.72E-04	1.10E-04	2.70E+00	1.75E-04
Subtotals	**1 .67E+01**	**3.99E+00**	**9.28E-01**		**2.75E-01**
Totals	**17.1**	**4.1**	**9.55E-01**		**3.2**

Bottom Core Barrel Section (41") Estimates

Total Surface Area	258 ft^2
Total Weight	9.48E+03 lb

Note: Sample data taken from Duke sample ID #Z10449 dated 5/15/1998

Table 2
Bottom Core Barrel Characterization Report

Component:	Bottom Core Barrel Section (41 Inches)		
Component Weight (lb.):		9.48E+03	
Total Activity Curies		1.37E+04	
Co-60 Activity Curies		7.17E+03	
Part 61 Table 1 C Fraction		0.31	
Part 61 Table 2 C Fraction		0.28	
Nuclide	**Curies/g**	**Estimated Curies**	**Scaling Factors**
C 14	3.15E-07	1.36E+00	1.89E-04
Mn 54	1.81E-05	7.81E+01	1.09E-02
Fe 55	1.26E-03	5.43E+03	7.58E-01
Co 60	1.67E-03	7.17E+03	1.00E+00
Ni 59	1.80E-06	7.73E+00	1.08E-03
Ni 63	2.41E-04	1.03E+03	1.44E-01
Nb 94	5.36E-09	2.31E-02	3.22E-06
Tc 99	1.19E-09	5.13E-03	7.16E-07
Totals	**3.19E-03**	**1.37E+04**	

ATTACHMENT 4

TRANSPORTATION and EMERGENCY RESPONSE PLAN

For

HADDAM NECK PLANT REACTOR VESSEL PROJECT

TRANSPLAN-9007

TRANSPORTATION and EMERGENCY RESPONSE PLAN

For

HADDAM NECK REACTOR VESSEL PROJECT

TRANSPLAN-9007

January 2000

Prepared for:

Bechtel Power Corporation

Prepared by:

WMG, Inc.
16 Bank Street
Peekskill, NY 10566

TRANSPORTATION AND EMERGENCY RESPONSE PLAN FOR CONNECTICUT YANKEE REACTOR VESSEL PROJECT REPORT TRANSPLAN-9007

WMG Inc.		16 Bank Street, Peekskill, NY 10566
Project Application	Copy No. NA	Assigned To: NA

APPROVALS

SIGNATURE(S) – DATE(S)

Rev No.	Preparer	Technical Reviewer	Approver
0	Al [illegible] - 3/9/00	Edward [illegible] 3/9/2000	[illegible] 3/9/00

FOREWORD

This report comprises Attachment 4 to the Exemption Request for the Connecticut Yankee (CY) Reactor Vessel Transport System. This work was performed under Subcontract 24265-TSC-200.

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1.0 SCOPE

1.1 Purpose

The purpose of this Plan is to provide comprehensive management, coordination, and control of the shipment of the Reactor Vessel Canister (Canister) from Connecticut Yankee Atomic Power Company (CYAPCO) Haddam Nuclear Plant (HNP) to the Chem-Nuclear Systems operated Waste Management Facility, located in Barnwell, South Carolina (Barnwell Site). The Plan also contains instructions for emergency response during land and water transit between HNP and the Barnwell site.

1.1.1 Applicability

This document is applicable to all CYAPCO/Bechtel personnel and subcontractors who are involved in the transportation of the Canister. The Canister will be prepared for off site transportation at HNP per References 2.11 and 2.12. The transportation of the Canister will commence upon its departure from the HNP's local barge slip and end upon the placement of the Canister in the disposal trench at Barnwell.

2.0 REFERENCES

2.1 U.S. Department of Transportation Exemption Request for Shipment of the Haddam Neck Plant Reactor Vessel Transport System

2.2 Title 49, Code of Federal Regulations

2.3 SC Department of Health and Environmental Control (DHEC) 61-83 Regulation

2.4 Attending Marine Surveyor's Report for CYAPCO HNP for RPVP

2.5 Bigge Crane and Rigging Co. Quality Assurance Manual, Project BCR-99-200

2.6 CYAPCO Radioactive Waste Shipping Procedures

2.7 46 CFR Part 15 Manning Requirement, Section 15.705 Watches

2.8 Bechtel Project QA Program Plan for the CY Decommissioning Project

2.9 Bigge Procedure BCR-PROC-2002-101, "Project Execution Plan"

2.10 ANSI 14.24

2.11 Bigge Procedure BCR-PROC-2002-103, "Tie-Down Package to Barge CYNP"

2.12 Bigge Procedure BCR-PROC-2002-104, "Transport to Barnwell Site"

2.13 Bigge Procedure BCR-PROC-2002-104.1, "Barge Transport to SRS"

2.14 Bigge Procedure BCR-PROC-2002-104.2, "Load and Tie Down to Transporter'

2.15 Bigge Procedure BCR-PROC-2002-104.3, "Land Transport to Barnwell Site"

2.16 Bigge Procedure BCR-PROC-2002-104.4, "Offload and Set in Trench"

2.17 Bigge Procedure BCR-PROC-2002-105. "Emergency Response Plan"

3.0 RESPONSIBILITIES

Overall project responsibilities and staffing are set forth in Reference 2.9. Responsibilities relating to the transportation of the Canister are summarized below.

3.1 Connecticut Yankee Atomic Power Company (CYAPCO)

CYAPCO, as the licensed operator of HNP, has contracted Bechtel as the Decommissioning Operations Contractor, which includes the offsite transportation of the Canister. The DOT exemption is issued to CYAPCO, and for the purposes of this plan, CYAPCO is the DOT shipper of record.

3.2 Bechtel Power Corporation (Bechtel)

Bechtel is under contract to CYAPCO as the Decommissioning Operations Contractor and has the overall responsibility for managing the decommissioning of HNP. Specific to this plan, Bechtel has the responsibility for the preparation and transport of the Canister in compliance with the DOT exemption.

3.2.1 The Bigge Project Manager (under subcontract to Bechtel) is responsible for the overall execution of off-site land and water transportation of the Canister. The Project Manager will coordinate the efforts of the Marine Surveyor, the Bigge Superintendents, and the Savannah River Site (SRS) Project Manager. The Project Manager shall also ensure adherence to the industrial safety and radiological standards of References 2.2, 2.3 and 2.6, carrier requirements of References 2.7 and 2.10, and support emergency actions as may be required by Reference 2.17 and Section 8.0 of this plan

3.2.2 The Bigge Health Physics Technician (BHPT) is responsible for providing health physics surveillance during transportation of the Canister and for responding to radiological emergencies. The BHPT will maintain the principles of ALARA per Reference 2.6 and radiological controls per Reference 2.3. This includes providing radiological training indoctrination, maintaining the dosimetry program, and conducting radiation and contamination surveys.

3.3 Bigge Crane & Rigging (Bigge)

Bigge Crane & Rigging, under contract to Bechtel, will provide heavy haul and support services for the off-site transportation of the Canister. This includes, barge and tug service, temporary bridge and loading ramp installation/removal, and crane/rigging services at the SRS and at the Barnwell sites. Bigge will also provide heavy haul and rigging services, to transport the Canister at the SRS and the Barnwell sit&. Bigge will supply a hydraulic platform trailer, prime mover and barge off-load rigging equipment. Bigge is designated as the carrier for the shipment of the Canister.

3.3.1 The Bigge Superintendent, or designee, is responsible for supervision of its crews for compliance to the requirements in this Plan and adherence to . ALARA and industrial safety requirements. The Superintendent is also responsible for implementation of the Emergency Response Plan per Section 8.0

3.3.2 The licensed tugboat Captain is responsible for the safety of the barge and crew under his direction, compliance with applicable maritime regulations, directing immediate actions in the event of an emergency per . Section 8.0, and supervision of the tug crew to fulfill requirements of this plan and all applicable regulations.

3.4 Marine Surveyor

A registered Marine Surveyor, under contract to Bigge, will provide an attending Surveyor to perform marine surveys of the barge and tug(s). This includes performance of pre-voyage (Pre-Tow) survey. of the barge and tug(s) several weeks in advance of the shipment to determine equipment adequacy and condition. Results of this survey will be provided to Bigge for corrective action. Prior to the shipment leaving HNP, the Attending Surveyor will inspect the barge tie-down arrangement for the Canister, and other sea-fasteners as may be required by the Attending Surveyor. The Attending Surveyor will also verify that the tugboat(s) and barge meet the requirements of the Pre-Tow and review the Trip-In-Tow recommendations with the tugboat Captain(s).

4.0 GENERAL REQUIREMENTS

4.1 The Bigge Project Manager (or his designee) shall indicate completion of the requirements in Sections 5.2, 5.3, 6.2, 6.3, 7.2, and 7.3 by initialing and dating in the spaces provided. The BHPT is authorized to sign off steps in Section 6.3 since these occur during the water transit.

4.2 Bigge's Quality Assurance Program, having been approved by Bechtel, will be implemented to assure items and activities meet or exceed established requirements. QA/QC requirements are contained in Reference 2.5.

4.3 One barge shipment will be made. The Canister will be tied-down to the barge.

4.4 The BHPT will accompany the shipment continually during the water transit. Health Physics coverage will also be provided during land transportation between SRS and the Barnwell site.

4.5 Industrial safety requirements shall be followed by all Bigge employees and subcontractors per Reference 2.2.

4.6 Due to the complexity of operations described in this Plan, it may be necessary to perform steps within a section concurrently or out of sequence. Should this become necessary, the Bigge Project Manager shall seek concurrence from the CYAPCO/Bechtel Site Coordinator (for work performed at HNP only) and the appropriate Superintendent(s) for all applicable work.

5.0 BIGGE PREPARATION AT HNP

5.1 Special Instructions/Precautions

(None)

5.2 Prerequisites

Prior to acceptance of the Canister on the barge, the following prerequisites shall be completed and verified per Section 4.1.

5.2.1 CYAPCO's exemption request from US DOT has been approved and exemption certificate issued.

____/____

5.2.2 CYAPCO has received South Carolina Department of Health and Environmental Control (DHEC) approval for the transportation and disposal of the Canister.

____/____

5.2.3 The barge is prepared to receive the Canister per Reference 2.11, and materials and personnel are available at the barge slip for collection/recovery of oil that may leak from the hydraulic mechanisms associated with the transportation equipment.

____/____

5.2.4 Acceptable weather conditions are forecasted for transport of the Canister.

____/____

5.2.5 Adequate daylight exists to complete the transport, barge loading and installation of a minimum number of barge tie-downs, as specified by the attending surveyor unless provisions for portable lighting are available.

____/____

5.2.6 Arrangements have been made for the marine survey of the tug(s) and loaded barge at HNP.

____/____

5.2.7 Bechtel has completed radiation worker training for Bechtel and Bigge personnel who require monitoring under its program during barge loading and tie-down installation.

_____/_____

5.2.8 The following courtesy notifications have been completed.

5.2.8.1 CYAPCO: To the town offices for each town along the Connecticut River that were formally in the Connecticut Yankee EPZ.

_____/_____

5.2.8.2 Bigge: To the States of South Carolina and Georgia, at least 10 working days prior to the scheduled departure date of the shipment from HNP.

_____/_____

5.2.8.3 Bigge: To the States along the Intercoastal Waterway route, at least 10 working days prior to the scheduled departure date of the shipment from HNP.

_____/_____

5.3 Barge Preparation

The following items shall be completed and verified per Section 4.1.

5.3.1 The Bigge Project Manager, or designee, shall conduct a pre-job briefing with personnel involved in the Canister loading, and land transporter tie-down installation.

____

5.3.2 Ensure that the land transporter tie-down system is properly secured to the barge per Reference 2.11.

_____/_____

6.0 WATER TRANSPORATION

6.1 Special Instructions/Precautions

6.1.1 Various tugs will be used during the voyage to meet the varying draft and duty requirements. The first tug(s) will transport the barge from the discharge canal at HNP to the Connecticut River. A second tug(s) will transport the barge from the Connecticut River to the inlet for Savannah River per Appendix A-l. A third tug(s) may transport the barge from the inlet for the Savannah River to the SRS boat ramp. All tugs will comply with the recommendations of Reference 2.10. Each set of tugs will consist of two tugboats; a primary tug and an assist tug. Each tug shall be manned as required per References 2.12 and 2.13 to support 24-hour vessel operation.

6.1.2 A primary and secondary means of communication shall be available between tugs and their base stations.

6.1.3 During the voyage, the tug(s) shall communicate to the Bigge Communication Center and CNS Barnwell Site Security, the shipment position a minimum of once-every 8 hours.

6.1.4 During the voyage, the tug(s) shall communicate to the Bigge Communication Center, CNS Barnwell Site Security, and the Connecticut Yankee Power Control room the following:

- Any scheduled or unscheduled layovers or delays exceeding 2 hours and
- Any unusual events, accidents, or emergencies affecting the shipment.

6.1.5 An emergency hawser will be provided on the barge and standard U.S. Coast Guard procedures for deployment utilized during the barge transport.

6.1.6 Speed will be limited to USCG and/or ANSI N14.24 applicable restrictions and surveyor sailing instructions.

6.1.7 Radar and navigational aides are operational on all vessels.

6.1.8 Except during the emergency situations described in Section 8.0, transportation of the Canister shall be direct and uninterrupted, following the route specified in Appendix A-l.

6.1.9 Prior to departure from any point along the route, the tug Captain is to assure that an acceptable weather forecast exists for the intended route. Acceptable weather forecast is per applicable USCG and/or ANSI N14.24. Should unpredicted conditions prevail, the shipment shall seek shelter per Section 8.0

6.1.10 The tug Captains shall comply with References 2.13 and 2.14 as prepared by Bigge with comments from the Attending Surveyor. Should conflicts arise between the requirements of this Plan and those prescribed in References 2.13 and 2.14, those in References 2.13 and 2.14 shall take precedence.

6.1.11 In the event an accident or other circumstance that results in the partial or total sinking of the barge, salvage operations will be employed. Bigge will immediately contact a Salvage Surveyor and Salvage Master to undertake required salvage operations. The first priority of salvage operations will be to mitigate any damage to and initiate recovery of the Canister.

6.1.12 Refer to Section 8.0 in the event of an emergency.

6.1.13 Each member of the tug crew shall be trained by Bigge. Training shall consist of radiation worker training and Haz Mat Employee training per 49 CFR 172, Subpart J. The assist boat crew, if required, is exempted from this requirement.

6.1.14 Tug(s) and/or assist boat(s) used to maneuver the barge in the HNP discharge canal shall be of appropriate size and horsepower to safely transport the barge under the prevailing weather and water conditions, as determined by the Attending Surveyor.

6.2 Prerequisites

Prior to the barge leaving the HNP barge slip, the following prerequisites shall be completed by Bigge .

6.2.1 The Canister has been prepared for off site transportation and land transporter system secured to the barge per Reference 2.11.

____/____

6.2.2 CYAPCO's shipping documentation, surveys, inspections and notifications have been completed and distributed per Reference 2.8.

____/____

6.2.3 Permit between DOE-SRS and Bechtel authorizing use of SRS facilities and highways is in place.

____/____

6.2.4 Pre-approval for overweight and oversize permits have been obtained from South Carolina DOT.

____/____

6.2.5 Permit application submitted to CSX Rail Road requesting rail crossing at SRS is approved.

____

6.2.6 The barge and tug(s) meet the Pre-Tow Recommendations of Reference. 2.10 as determined by the Attending Surveyor.

____

6.2.7 The BHPT and Bechtel HPT has performed and documented a radiation and contamination survey of the barge.

____

6.2.8 A Dangerous Cargo Manifest, prepared by Bechtel, has been signed by the tug Captain.

____

6.2.9 The following documents have been provided to each tug Captain and a copy is secured in a weatherproof cover on the barge.

____/____

- Shipping paperwork package prepared by CYAPCO,
- Dangerous Cargo Manifest,
- Bechtel radiological survey,
- References 2.12 and 2.13, and
- Copy of this Plan.

6.2.10 Each tug has sufficient fuel- to complete the planned portions of the voyage.

____/____

6.2.11 Tug boat(s) and crew,. as prescribed in the preceding sections are available to receive the shipment at the HNP Discharge Canal.

____

6.2.12 Arrangements have been made for shallow draft tug service out of Savannah, Georgia

____

NOTE: TRANSPORTATION SHALL PROCEED ON THE CONNECTICUT RIVER AND SAVANNAH RIVER ONLY DURING DAYLIGHT HOURS.

6.2.13 BHPT, equipped with dosimetry and radiation monitoring equipment, is aboard the primary or assist tug and has completed indoctrination of the tug crew.

____/____

6.2.14 Bigge preparation of the SRS boat ramp is complete.

____/____

6.2.15 Primary and backup means of communication between tug and base stations are satisfactorily tested.

____/____

6.3 Transportation

The following items shall be completed and verified per Section 4.1:

6.3.1 The primary tug Captain verifies an acceptable weather forecast.

____/____

6.3.2 Under the direction of the tug Captain, unsecure barge mooring lines and remove barge from the slip.

____/____

6.3.3 Tug Captain shall notify the nearest Captain of the Port and Bigge Communications Center of the shipment's departure.

____/____

6.3.4 At the Connecticut River, make up the tug to the barge.

____/____

6.3.5 The shipment will proceed along the route depicted in Appendix A-2.

____/____

6.3.6 At Savannah, GA, the ocean tug(s) will be replaced with tug(s) prescribed in References 2.12 and 2.13 for transit on the Savannah River.

____/____

6.3.7 Notify SRS a minimum of four (4) hours in advance of the barge's scheduled arrival at the SRS boat ramp.

____/____

6.3.8 Upon arrival at the SRS boat ramp, moor the barge to the dock area adjacent to the boat ramp.

____/____

6.3.9 Conduct an arrival radiation and contamination survey of the shipment and establish radiological postings at the gangway.

____/____

6.3.10 Position the barge in boat ramp, moor and make barge ready for off load.

____/____

7.0 LAND TRANSPORTATION AT SRS AND BARNWELL

7.1 Special Instructions/Precautions

7.1.1 Escorts and traffic control will be provided along the haul route to isolate the shipment from commercial vehicle traffic and personnel not directly involved in the shipment and to allow the shipment to proceed without delay or interference.

7.1.2 At SRS, shipment escort will be provided by DOE and Bigge. On State of South Carolina roads, escort will be provided by DOE Security and local law enforcement agencies. Bigge will provide personnel, equipment and materials for traffic control. Traffic control personnel will be under the direction of Bigge/DOE Security.

7.1.3 Speed shall be limited to 5 mph or less.

7.1.4 Bridges, culverts or underground utilities along the route identified by SRS or SC Highway authorities as being structurally inadequate will be spanned using transition beams and/or steel plates.

7.1.5 The prime mover may be parked overnight in one of the designated safe berths along the route. During periods when the prime mover is parked in a safe berth;

- Portable lighting will be established to illuminate the safe berth area,
- Radiological postings will be established around the Canister, and
- Periodic surveillance by SRS Security of the safe berth will be conducted or if parked off SRS property, a watch shall be posted at the safe berth that is equipped with mobile communications.

7.1.6 Refer to Reference 2.17 and Section 8.0 in the event of an emergency.

7.2 Prerequisites

Prior to the shipment leaving the SRS boat landing area, the following prerequisites shall be completed and verified per Section 4.1.

7.2.1 Confirm with the Barnwell Site Manager that the site will be prepared to receive the shipment on the anticipated arrival date.

____/____

7.2.2 Verify that the canister surface temperature is above 0°F before commencing off-load of the canister from the barge to the land transporter.

____/____

7.2.3 Verify that the Canister is properly secured to the land transporter as per Reference 2.14.

____/____

7.2.4 Verify that traffic control personnel, equipment and supplies are ready to, support the movement of the prime mover (including hydraulic oil spill recovery) and that designated safe berths are available.

____/____

7.2.5 Verify that CSX personnel are available at the rail crossing on SRS Road 3 to authorize shipment across the tracks.

____/____

7.2.6 Notifications have been made to local (Barnwell) law enforcement agencies and utility companies of the shipment scheduled departure time from SRS.

____/____

7.2.7 The physical conditions of road and bridge surfaces along the route have been visually documented by Bigge, as necessary.

____

7.2.8 The haul route has been surveyed for potential interference from structures such as highway postings, poles etc. and resources are available to remove and re-install them as the shipment proceeds.

____/____

7.2.9 **Bigge** personnel, equipment, and materials are staged to span bridges, culverts, and underground utilities, as required, along the route.

____/____

7.2.10 Perform **pre-departure** inspection of prime mover and transporter per Reference 2.15.

____/____

7.2.11 Original shipping paperwork is placed in the cab of the prime mover.

____

7.2.12 **Re-inspect** the required DOT required labels, markings and placarding.

____

7.3 Transportation

The following items shall be completed and verified per Section- 4.1.

7.3.1 The shipment will proceed along the following route as depicted in Appendix A-2:

River access road to River Road (A-4), South on River Road (A-4) to Road 3, Road 3 through Barricade 7 to Road 5, South on Road 5 to Road 6, East on Road 6 to Road F, South on Road F to Road B, East on Road B to Barricade 4 (Snelling Barricade). From the Snelling Barricade, the shipment will proceed East on SC 64, to the gravel road connecting SC 64 to Osborne Road. Take the gravel road to Osborne Rd, and then proceed to the main gate of the Barnwell site

____/____

7.3.2 Upon arrival at the Barnwell site, the shipment will be inspected and received per References 2.15 and 2.16.

__________/Chem-Nuclear

8.0 EMERGENCY RESPONSE PLAN

This section is developed to provide instructions for comprehensive notifications, reporting, management, and emergency actions in response to an emergency occurring during the voyage. This section shall be used as a guideline.

NOTE: REFER TO REFERENCE 2.17 FOR HIGHWAY EMERGENCIES.

8.1 Responsibility

8.1.1 Emergency Coordinator: During barge transit, the ocean or river tug Captain shall be designated the Emergency Coordinator who is responsible for the implementation of the emergency plan. All classifications, notifications, assignments, and follow-up actions shall be delineated by the Emergency Coordinator.

8.1.2 Radiation Advisor: The BHPT shall coordinate and provide recommendations on any radiological events.

8.1.3 Maintenance Supervisor: The Bigge Superintendent shall coordinate, recommend, and implement mitigating actions to ensure the safe operation of the barge support equipment.

8.1.4 Communicator: The Bigge Communication Center operator is responsible for completing the Emergency Notification Incident Form in Appendix B, taking further actions, and completing notifications per Appendix C.

8.2 Emergency Classifications

NOTE: MORE THAN ONE CLASSIFICATION MAY EXIST AS A RESULT OF AN ACCIDENT. THE GOAL OF THE CLASSIFICATION IS TO BRIEFLY CATEGORIZE AND IDENTIFY POTENTIAL RESPONSES FOR ANY EVENT.

8.2.1 Weather: All weather that threatens the ability of the Captain and crew to safely transport the barge and/or the tug(s).

8.2.2 Communications: All events whereby all communications are lost between tug(s) and the shore.

8.2.3 Tie-Down Equipment: Any malfunctions of the in-place tie-down equipment that jeopardizes its ability to secure the Canister.

8.2.4 Tugboat: All equipment failures that jeopardize the ability of the in-place tug(s) to control the barge.

8.2.5 Radiation Protection: Any significant increase in surface contamination or dose rate at the Canister boundary.

8.2.6 Grounding/Collision: Physical contact made with another obstacle

8.3 Emergency Actions

NOTE: ALL ACTIONS SHALL BE MADE WITH THE FOLLOWING PRIORITIES:

- **PUBLIC SAFETY**
- **PERSONNEL SAFETY**
- **TRANSPORT PROTECTION**

8.3.1 Once an emergency is identified, the Emergency Coordinator shall delegate any immediate actions that may preclude further problems.

NOTE: THE REMAINING ACTIONS ARE BASED UPON THE CLASSIFICATION OF INCIDENT.

8.3.2 Once the immediate actions are delegated, the notifications identified in Appendix C shall be made by the Emergency Coordinator. The Emergency Coordinator may delegate this responsibility to the Bigge Communication Center. The Communicator shall complete an Emergency Notification Incident Form, Appendix B, and any actions as directed by the Emergency Coordinator.

8.3.3 Weather

8.3.3.1 Seek the closest safe harbor and secure barge.

8.3.3.2 Once secured, the Radiation Advisor shall restrict access surrounding the barge based upon the radiological conditions around the barge.

8.3.3.3 Once the weather has passed and the near term weather conditions are satisfactory, reinitiate transport.

8.3.4 Communications

8.3.4.1 Seek the closest safe harbor and secure barge.

8.3.4.2 Once secured, the Radiation Advisor shall restrict access surrounding the barge based upon the radiological conditions around the barge.

8.3.4.3 Obtain/repair communication equipment.

8.3.4.4 Once communications are restored, reinitiate transport.

8.3.5 Tie-Down Equipment ,

8.3.5.1 Communicate with appropriate Bigge/Bechtel personnel to assess the significance of the problem.

8.3.5.2 In the event the tie-down equipment damage jeopardizes the security of the barge, seek the closest safe harbor. Also, attempt to install temporary rigging.

8.3.5.3 Once secured, the Radiation Advisor shall' restrict access surrounding the barge based upon the radiological conditions around the barge.

8.3.5.4 Root cause of the failure shall be determined by Bigge personnel and addressed prior to reinitiating transport.

8.3.5.5 Once rigging has been fixed and/or modified, reinitiate transport.

8.3.6 Tugboat

8.3.6.1 Replace non-functioning tug and await necessary support. If possible, perform necessary repairs.

8.3.6.2 Once repaired and/or replaced reinitiate transport.

8.3.7 Radiation Protection

NOTE: DURING TRANSIT, SURVEYS WILL NOT BE TAKEN. HOWEVER, IF THERE IS CAUSE FOR RADIOLOGICAL CONCERN, A SURVEY SHOULD BE PERFORMED.

8.3.7.1 Identify the source of contamination by performing radiological surveys.

8.3.7.2 If the containment structure is breached, attempt to isolate it. Restrict access as required to maintain appropriate radiological controls.

8.3.7.3 Based upon discussions with Bigge and CYAPCO/Bechtel personnel, either seek shelter or repair the structure to complete transport.

8.3.7.4 The Radiation Advisor shall ensure the necessary radiological controls are maintained.

8.3.8 Grounding, Collision, Etc.

8.3.8.1 Request the necessary support equipment and personnel based upon discussions with Bigge and CYAPCO/Bechtel.

8.3.8.2 In the event a catastrophic emergency was to occur, Bigge would establish an on-shore Emergency Operations Facility. This Facility shall coordinate the salvage, repair, etc. The responsibility of such actions would be established under Reference 2.17.

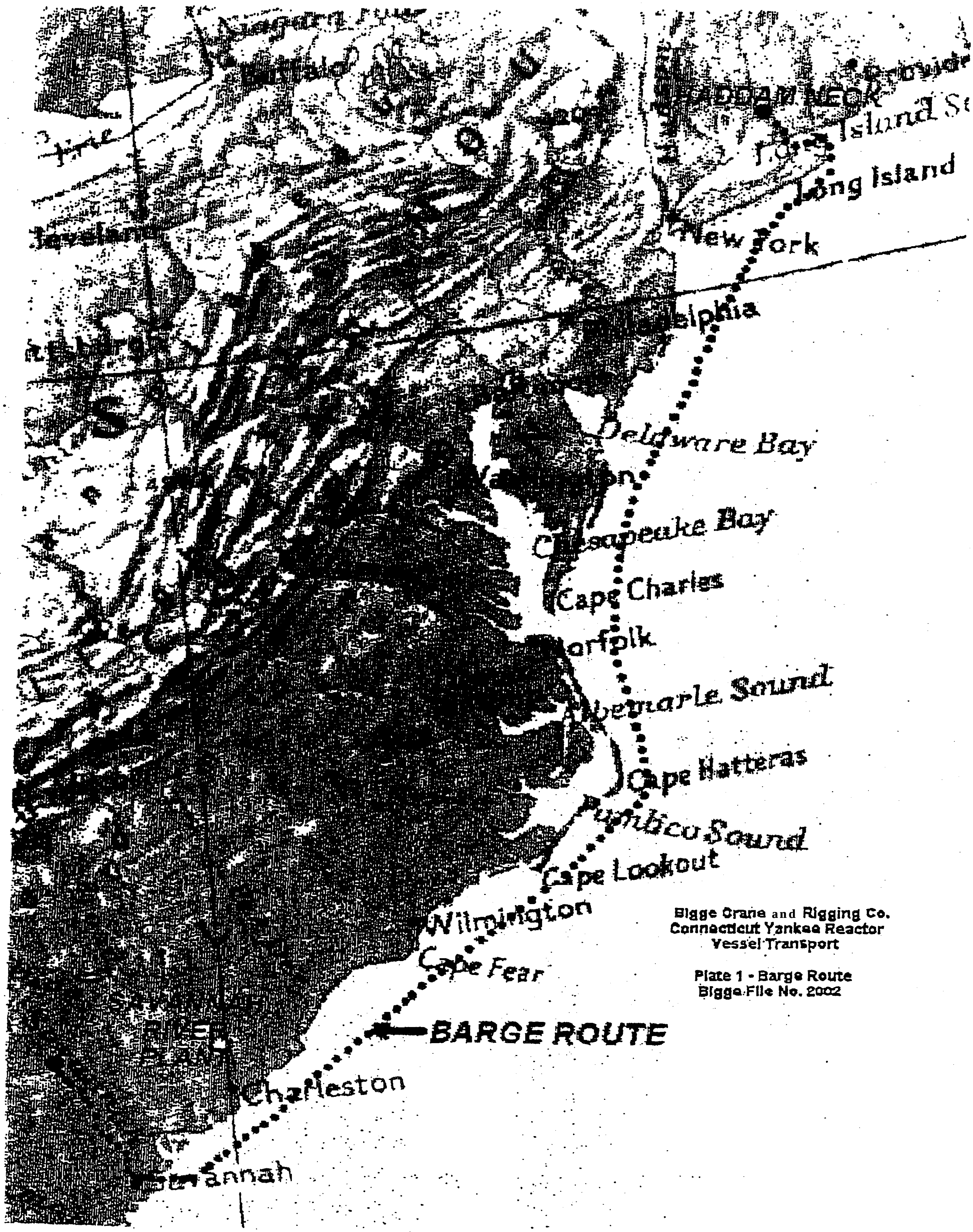
9.0 RECORDS

Quality Assurance records generated as result of this Plan shall be maintained in accordance with References 2.5 and 2.8.

APPENDIX A-l

WATER TRANSPORT

(1 PAGE)



Bigge Crane and Rigging Co.
Connecticut Yankee Reactor
Vessel Transport

Plate 1 - Barge Route
Bigge File No. 2002

APPENDIX A-2

LAND TRANSPORT

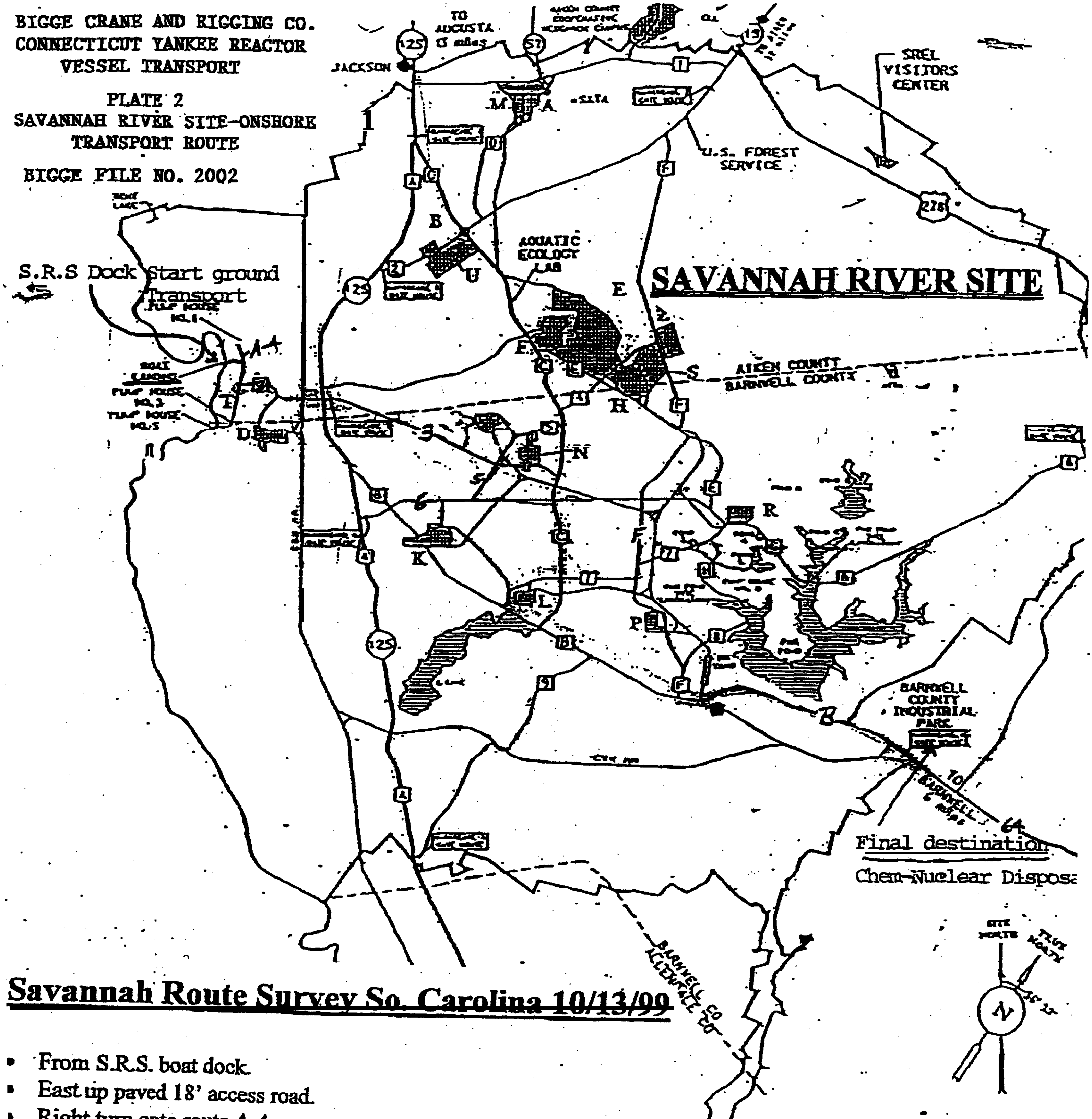
FROM SRS TO BARNWELL

(1 PAGE)

BIGGE CRANE AND RIGGING CO.
CONNECTICUT YANKEE REACTOR
VESSEL TRANSPORT

PLATE 2
SAVANNAH RIVER SITE-ONSHORE
TRANSPORT ROUTE

BIGGE FILE NO. 2002



Savannah Route Survey So. Carolina 10/13/99

- From S.R.S. boat dock.
- East up paved 18' access road.
- Right turn onto route A-4.
- Left turn at route 3..... cross over State route 125
- Right turn at route 5.
- Left turn at route 6.
- Right turn at route F.......11.5 miles from river site
- Left turn at route B .....drive out east gate & enter State Route 64 for apx. 200 feet.
- Left turn onto gravel access road to Chem-Nuclear Systems.
- Enter disposal site at far right gate.

Total mileage of route: 22.9 miles.

APPENDIX B

EMERGENCY NOTIFICATION INCIDENT FORM

(1 PAGE)

EMERGENCY NOTIFICATION INCIDENT FORM

LOCATION: Latitude: ______________ Longitude: ______________

PROBLEM: Classification: __________

Description: ____

____

____

____

____

Action(s) Taken: ____

____

____

____

Notifications: ____

____

____

Corrective Actions:
To Be Taken: ____

____

____

____

Radiological
Assessment: ____

____

____

____

Communicator: ________________________ **Date/Time:** __________________

APPENDIX C

EMERGENCY NOTIFICATION LIST

(1 PAGE)

EMERGENCY NOTIFICATION LIST

1.	Bigge – Fred McFall, Project Sponsor	Office: (206) 443-8500 Cell: (206) 321-4779 Home: (206) 236-0483
2.	Bigge – Jim O'Callaghan, Project Manager	Office: (510) 638-8100 Cell: (510) 918-8901 Home: (415) 585-6097
3.	Coast Guard	(Radio)
4.	Chem-Nuclear System Security	(803)-259-6069
5.	Connecticut Yankee Power Company Control Room	(860)-267-3211
6.	Robert Vallem, Bechtel Waste Management Supervisor	(860)-267-3063 (800)-334-1391 Pager

CONNECTICUT YANKEE ATOMIC POWER COMPANY

HADDAM NECK PLANT
362 INJUN HOLLOW ROAD. EAST HAMPTON, CT 06424-3099

March 30, 2000
CY-00-016

Ref. 49CFR107 & 49CFR173

Associate Administrator for Hazardous Materials Safety
Research and Special Programs Administration
U.S. Department of Transportation
400 7th Street, SW
Washington, D.C. 20590-0001

Attention: Exemptions, DHM-3 1

SUBJECT: REQUEST FOR EXEMPTION FOR SHIPMENT OF THE HADDAM NECK NUCLEAR PLANT REACTOR VESSEL

Connecticut Yankee Atomic Power Company (CYAPCO) is in the process of decommissioning the Haddam Neck Plant (HNP). Therefore, we hereby request the necessary exemptions for the shipment of the reactor vessel within a Reactor Vessel Transport System (RVTS) from the HNP to the low-level radioactive waste burial site at Barnwell, South Carolina. The RVTS will provide a package/transport system with an equivalent safety level to that of a DOT Industrial Package Type 2 (IP-2) package. The RVTS will be transported under a DOT exemption pursuant to 49 CFR Part 107.105 via barge to the Savannah River Site (SRS) in Aiken, South Carolina and via land transporter from SRS to Barnwell. This shipment will be exclusive use, one-time only, and will be performed in accordance with the transportation plan as described herein. The current project schedule reflects the shipment to occur after September 1, 2000.

The intact HNP Reactor Pressure Vessel (RPV) containing some reactor internals components and potentially RPV mirror insulation will be packaged and grouted, with low density cellular concrete, within a 3-inch thick steel canister with the RPV head bolted to the exterior of the canister. The components of the reactor vessel internals exceeding 10 CFR Part 61 Class C limits (Greater Than

Class C, GTCC) will be removed and will not be shipped with the RPV. The GTCC will remain at the HNP site for future disposition.

The RVTS consists of the "package" which is used to contain the RPV and its Class 7 (radioactive) materials, an integrated tie-down system for barge and land transport, and a transportation and emergency response plan. A complete description and discussion of the exemptions requested, the RVTS, its transportation plan, and emergency response plan is provided in the following four (4) attachments to this request:

Attachment 1 – Compliance Matrix
Attachment 2 – Transport System Description
Attachment 3 – Reactor Vessel and Internals Characterization
Attachment 4 – Transportation and Emergency Response Plan

The exemptions requested for one-time use of this transport system include that:

1. The package is designated as a non-specification package since the proposed Transport System, under normal conditions of transport and prescribed operating conditions, provides safety equivalence to that of an IP-2 package,

2. The package is exempted from the drop requirement from the orientation that causes ***"maximum damage",***

3. The package contents is considered LSA III material,

4. The package contents, including the RPV itself, is considered a ***"collection of solid objects"*** and that the requirements of 10 mSv/hr (1R/hr) at 3 meters, as provided in 49 CFR Part 173.427(a)(1), be applied from the unshielded surface of the RPV exterior,

5. The package contents classified as LSA-III materials is exempted from the leachability requirements of 49 CFR Part 173.468, and

6. The package is exempted from the stacking test requirements of 49 CFR Part 173.465(d).

The RVTS will meet all of the other requirements of 49 CFR Part 173.

The bases for the requested exemptions are due to the unique characteristics of the Class 7 (radioactive) material to be transported, the packaging, and the administrative controls that will be implemented during transportation. The justification for each exemption requested is provided in the Compliance Matrix (Attachment 1). The remaining attachments provide additional supporting information.

The Compliance Matrix (Attachment 1) addressing the requirements of 49 CFR Part 107 and Attachments 2 through 3 together demonstrate that the shipment of the HNP reactor vessel can be performed with a level of safety equal to or greater than that of an IP-2 package.

At your convenience, we are prepared to meet with you to discuss this request and respond to any questions. If additional information is required, please contact Mr. Gerry P. van Noordennen, Manager of Regulatory Affairs, at (860) 267-393 8.

Sincerely,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

Russell A. Mellor – Vice President
Operations and Decommissioning

Attachments:

Attachment 1 – Compliance Matrix
Attachment 2 – Transport System Description
Attachment 3 – Reactor Vessel and Internals Characterization
Attachment 4 – Transportation and Emergency Response Plan

cc: R. Boyle, DOT, Office of Hazardous Materials Technology
T.L. Fredrichs, Project Manager, USNRC